NUCLEAR ENGINEERING VOL. 6 NO. 64 SEPTEMBER, 1961

Fast Reactor Breakthrough

WE have been a shade doubtful in the past of the wisdom of regarding the fast breeder as the inevitable plutonium burner and not a little concerned at the policy of basing our whole plutonium burning programme on fast reactor technology. Even before the first flush of enthusiasm for the system had passed it seemed that the degradation of the spectrum resulting from the introduction of constructional materials and coolant would lead to a conversion ratio that was so close to unity that the term breeding (when taken to imply multiplication) ceased to be meaningful. At a time when the basic *raison d'être* of the fast breeder was to provide a means of burning U^{238} in order to conserve our uranium supplies this was a matter of some significance. It is less material now that uranium is in plentiful supply but then the chief function of the fast reactor becomes one of burning plutonium (or uranium) so as to produce power cheaper than by any other means, and the chronic problem of high fuel burn-up becomes paramount.

The litigation in the U.S.A. has focused attention on the safety problems of the fast reactor which tend to be grossly exaggerated in the lay mind as the terms fast and slow are so often taken as referring to the speeds of development of divergent reactions. This is, of course, not the case and whilst the prompt neutron lifetime in a fast reactor is much smaller than a thermal reactor and prompt criticality would therefore lead to a much faster divergence, prompt criticality is a condition to be avoided in any reactor be it fast or thermal. Of more importance is the smaller delayed neutron fraction which implies there is a much smaller margin between delayed criticality and prompt criticality, but this is compensated by the fact that a much smaller total reactivity need be invested as the poisoning effects of fission products and the temperature coefficients are so much less. Whilst it is undoubtedly true that one can postulate a more serious nuclear incident with a fast reactor than with a thermal—simply by assuming that a fuel meltdown and agglomeration is possible—this is not to say that the probability of a nuclear incident is any higher than with any other highly rated system. There is of course the additional complication of a sodium-water chemical reaction, but we suspect that the seriousness of this problem has also been rather exaggerated and the precautions, at present felt necessary, grossly extravagant when compared with an ordinary chemical plant where comparable conditions prevail.

Nevertheless the fast reactor, because of the consequences of both prompt criticality and a sodium-water reaction, is likely to be much more expensive to build than its compact appearance would, on the face of it, warrant. This will be true at least until much greater experience has been gained in the operation of such plants as EBR-2, Dounreay, Enrico Fermi and the Russian BR50. On the other hand when reasonable engineering solutions become generally acceptable and the alarmist hypothesizer has not the same ascendency as today, it is reasonable to hope that the capital cost will come down appreciably. In his article on "An Economic Fast Reactor" (p. 371) Dr. Blake in fact goes further than this and claims that it will be lower than for any other system.

If there were only Dounreay to extrapolate from, this view would demand an excessive amount of faith but, fortunately, international collaboration on fast reactor technology is good and everyone interested can draw upon the experience of the designers and builders of fast reactors throughout the world. One might go further and say that if we were working in isolation, then the design of the DFR was rather disastrous as so many decisions, taken in good faith at the time when the DFR was on the drawing board, have since been proved unfortunate, for example, the frantic complexity of the primary circuit and reactor vessel and the inadequate provision for sodium clean-up. In the broad context, however, of the international group of fast reactors the DFR will be able to make a real contribution to fast reactor experimentation.

Enrico Fermi, on the other hand, while by no means representing the ultimate in fast breeder design, has demonstrated that the many novel problems are amenable to a conventional engineering approach and PRDC are to be congratulated on the neatness and cleanness of their finished product. We have yet to see whether there are any major snags still to be overcome but the background testing and plant commissioning so far completed give grounds for optimism. We trust that the company will be allowed to run their reactor with the minimum amount of interference (a pious hope that we see little chance of being fulfilled).

One should say immediately that, whatever its deficiencies, DFR has the great advantage of having actually run although just at this moment it is shut down while modifications are made to the system to prevent gas entrainment. Chief sources for the ingress of gas were the expansion tanks where vortices formed when the level dropped and exposed the downcomers, the thermocouple tubes and the spaces between the control rod tubes and outer sheaths. But the necessary modifications should have been completed by the end of August and the reactor will then be taken up to power reaching 15-20 MW by next spring after which a new core will be installed.

While, then, one could be reasonably confident that provided the right sort of research and development were undertaken and provided the various safety authorities could be persuaded to adopt a reasonable attitude capital costs would also become reasonable, fuel element burn-up still remained a near-insuperable stumbling block. With

burn-ups limited to less than 1% the chance of the fast reactor becoming economic seemed remote, even if the price of plutonium were much lower than the present quoted £6/g. We felt that whilst you could design out of distortion troubles, heat exchanger complexities, gas entrainment disturbances and so on, there seemed little chance of designing out of this central burn-up problem. In this it now seems pretty certain we were wrong and the experiments conducted in the DMTR at Dounreay and reported on page 373, in which both metal and oxide fuel elements have been taken to 5% burn-up without damage would seem to prove conclusively that such high burn-up figures can be achieved in a practical power reactor and with a reasonably low element fabrication cost.

The key to the high burn-up has been the introduction of a void in the fuel and the containment of the fuel within a strong can able to withstand the pressure build-up of gaseous fission products. With the metal, the change in shape of the fuel which occurs on irradiation resulting in a dense outer volume tightly bonded to the can and a porous centre section where fission products are retained is the sort of fortunate effect which one has rather ceased to expect; lately it has seemed that when one considers the effects of irradiation on materials, always one's worst fears are realized. Here the opposite seems to be true and the voided fuel element seems to be the breakthrough that reactor technology and the fast reactor in particular was needing.

The odd thing is that the U.K.AEA seems singularly unexcited about this work and when questioned about fast reactor fuel element policy the answers today are similar to those given 12 months ago, viz.: that most attention is being paid to a mixed oxide stainless steel cermet with some interest in carbide which might also be used in a simple pin-type weak-can element. There is, of course, bound to be some hesitancy in swinging over to a new policy and a great deal of confirmatory work needs to be put in hand to ensure that these initial results are not simply a fluke but are reproducible. There is also, of course, a great deal of allied work necessary on, for example, corrosion before the specification of a fuel element can be determined, but this is the sort of programme that can be measured in months rather than numbers of years which is the time scale of the present fuel element programme.

There is every indication that PRDC are keenly interested in the work on the voided fuel and it seems likely that they will initiate in the not too distant future their own research programme based on the Dounreay experiments. With so much effort being required in the completion of the Enrico Fermi plant and the preparation of the enormous quantities of paper work required for safety assessments and the like, PRDC could be forgiven for treating the U.K. work with a certain off-handedness so it is doubly strange that they should in fact be sitting up and taking notice while the AEA remains unimpressed.

Can it be that the one half is ignorant of the activities of the other or is it simply a little coyness in case of setbacks later. If snags are expected then let us hear of them now and let us see experiments begun immediately which will prove unequivocally the qualities of the voided fuel. If we are eventually to reject it let us reject it for the right reasons and not simply let it go by the board because nobody got around to changing the established order. If the DFR core is to be changed next spring let us see a voided metallic fuel in a strong can installed; there is plenty of time. This could be the big chance the nuclear engineering business has been needing.

U.K. and the Common Market

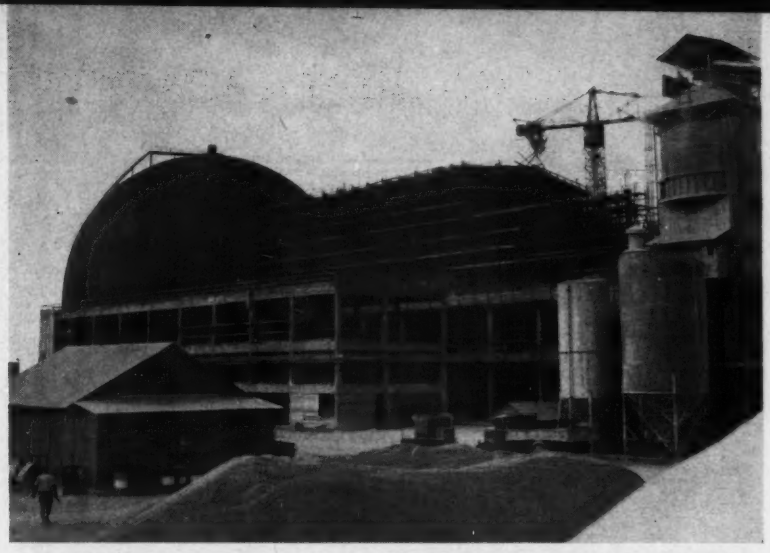
THE United Kingdom has formally notified the Common Market that it would like to begin negotiations on integration with the Six. Discussions are likely to be protracted and it is conceivable that they might end in failure but there is no virtue in assuming anything other than that they will be successful and as a result that the U.K. will also be joining Euratom. Now is the time then to begin preparations for this special merging of interests.

Last month we referred to the need for a definition of the areas of responsibility of the international atomic energy bodies and this clearly becomes even more necessary when the weight of the U.K. is added to that of the existing Six. There is still room for ENEA while the Scandinavian countries remain outside but it becomes a rather unbalanced group. OECD itself is unlikely to remain undisturbed and it may be that the functions of ENEA cannot be reviewed until the responsibilities of its parent are revised.

There are other matters that need serious consideration. When the U.K. becomes a member of the Common Market and European markets become open to us so do U.K. markets become open to European manufacturers who in the reactor field are building up a great deal more experience in width (although not in depth) than U.K. industry. By joining the Common Market the home market ceases to mean the CEGB but all the utility organizations within the Six as well. A progressive linking up of the Continental grid systems makes conditions there more comparable to those in the U.K. but it is already clear that the Calder reactor cannot be the universal panacea. Integration changes the fundamental precepts upon which our programme is based and a fundamental change in development policy should follow. We shall, of course, be able to climb on the backs of those who have been reaping the benefits of the Euratom-U.S. agreement in terms of scientific knowledge but as we were so anxious to point out three years ago this is quite distinct from commercial know-how which takes much longer to acquire.

Perhaps though this also must wait until more fundamental issues have been resolved, notably the future functions of the U.K.AEA vis-à-vis industry. The old order must change, as the policies established over the next months can make or mar the prospects of the U.K. nuclear industry for many years. Traditional ideas on patents rights, access agreements, raw material monopolies will have to change or we may find that the grip this country had on commercial nuclear construction has gone and, as happened in the shipbuilding industry, we slide from being a major exporter to being an importer.

World Digest



The 150MW BWR being built at the mouth of the River Garigliano, Italy, for Sann by General Electric. Euratom is to spend up to $7 million on the project and in return will obtain information on the reactor and its operation. (See page 361.)

International

The IAEA is now sending "Follow-Up Missions" to countries already visited by Agency preliminary assistance missions. Their purpose is to re-establish contacts made during the first visit, ascertain the progress made in the countries' nuclear programmes, to give advice on specific projects and determine the role the Agency could play in furthering these projects. The first countries to be visited by a "Follow-up Mission" are Burma, Thailand, Indonesia, the Philippines, China and Korea.

INCREASES in environmental radioactivity in OEEC countries have now to be reported to the appropriate authorities in neighbouring countries and the ENEA, under a scheme approved by the OEEC Council on July 7. Three levels of increased activity are covered by the scheme. These are designated as "emergency," when arising from a nuclear incident and of such a level to entail regional health hazards; "pre-emergency," for increases arising from activity of unknown origin or resulting from small incidents which do not call for local health protection measures; and "scientific interest," covering abnormal but slight increases which are of known or unknown origin and might be of interest to, for instance, meteorologists. Notification in the latter case is optional. At the moment the scheme applies only to atmospheric contamination and for guidance the values of the three levels are given in terms of concentration in air of total beta activity from artificial radionuclides, namely, 1 000 $\mu c/m^3$, 100 $\mu c/m^3$, and 10 times the previous monthly average level when exceeding the value of 5 $\mu c/m^3$, respectively. Later the scheme will be extended to take in other contaminable media such as river waters. ENEA worked out the scheme and after two years' operation the Agency's Steering Committee will report on it.

A supplementary protocol to the Polish-Soviet agreement for mutual co-operation in the utilization of nuclear energy has been signed. It provides for further co-operation in the fields of radioisotope applications in industry and medicine, radiation chemistry and nuclear physics.

About 250 papers have been accepted for the IAEA sponsored conference on plasma physics and controlled nuclear fusion research at Salzburg, Austria, September 4-9. Among them are about 80 from the U.S., 60 from the U.S.S.R. and between 20 and 30 each from the U.K., France and Germany.

Association with the Russians and Americans on a study into accelerators with energies in the 500-1 000 GeV range is being proposed by CERN, according to Mr. John B. Adams in the Organization's report for 1960. At the same time, Mr. Adams, who has now left Geneva to take up his duties as director of the U.K.AEA Culham laboratories on a full-time basis, urges CERN to make plans for having by 1970 a machine larger and better than anybody else's. Otherwise, he says, the Organization would again be in third place.

Over 120 nuclear scientists and engineers from 21 countries and two international organizations attended the IAEA-sponsored seminar on fast and intermediate reactors, Vienna, August 3-11.

Euratom is to take part in the Italian CNEN-Fiat-Ansaldo and Dutch RCN nuclear ship research and development programmes and in a general nuclear marine research programme being undertaken by GKSS of Hamburg. The latter participation is additional to Euratom's part in the GKSS-Interatom programme for which a contract was signed earlier this year ("World Digest," March, 1961) and covers investigations into vibration and shielding. The Italian project is centred on a water reactor for a tanker and is scheduled to last for two years. Euratom will contribute $1·2 million. RCN's programme involves the adaptation of a PWR for ship propulsion and will last for three years. The contribution from Euratom will be $1·9 million. To the German programme, to last five years, Euratom is prepared to contribute $1·8 million. Concurrent with the technical research projects Euratom will arrange for a comprehensive study of such questions as tonnage, ship type and voyage lengths for which nuclear-propelled ships are best suited.

Delegates from the 12 member countries of the Antarctic Treaty at a meeting in Canberra during July agreed to an exchange of information on nuclear equipment and techniques with special application to the Antarctic, as, for instance, the Soviet ice-breaker, *Lenin*, and the U.S. portable nuclear power plants. The Treaty, which was signed in Washington in 1959 and came into force on June 23 of this year, reserves the Antarctic area for peaceful purposes, forbidding all military activity there; provides for international co-operation in scientific investigation and research; and freezes the *status quo* with regard to territorial sovereignty, rights and claims. The signatories are Argentina, Australia, Belgium, Chile, France, Japan, New Zealand, Norway, South Africa, the Soviet Union, the U.K. and the U.S.A.

ENEA to Face Ship Problems

NO clear-cut proposals will be presented to the ENEA Ship Propulsion Study Group at its meeting on September 15 by the two working parties established to examine the economic and legal aspects of a nuclear ship, according to reports from Rome. The first party has apparently found that due to the lack of time and the incompleteness of available data it is impossible to state for certain whether it is possible or not to build a nuclear ship that would at present compare favourably from the economic point of view with a conventional vessel. However, if a ship is to be built in the near future then, according to the party's view, it should be based on either a water reactor or an OMR. The Rolls-Royce Vickers SCHWR (described in detail in *Nuclear Engineering*, June, 1961) was considered but it was decided it would be several years before a prototype would be possible. More interest was given to the Vulcain project submitted by BelgoNucléaire and it is likely that the party will re-examine this project in the light of changes to be undertaken by BelgoNucléaire and CEN (see item under "Belgium"). On the choice of ship to be installed with a reactor the British support a tanker, the Norwegians an ore carrier, and the French an oceanographic research vessel. As it is intended that any project embarked upon will be a joint enterprise the question of the flag under which the ship would sail has also to be considered. It is felt that to fly the flag of a particular participating State would be unsatisfactory. Alternatively, the adoption of a European flag would involve difficulties arising from international maritime law and necessitate the creation of new regulations. A compromise suggested by one of the working parties is that the ship should fly a national flag accompanied by a European flag as if it were a shipowner's flag.

United Kingdom

The 135 ft dia. containment building of the AGR was successfully pressure tested over the August Bank Holiday. Results from sub-atmospheric, strength and leak tests were stated to be very satisfactory. The building is designed to withstand internal pressures of 10 psi at which pressure the leakage rate should not exceed 0·1%/d of the net volume.

The DFR is expected to be started up again during September.

Tenders for Oldbury and subsequent CEGB nuclear power stations are to be paid for by the Board. Each tender submitted, whether selected or not, will be paid for at the rate of £100 000. The scheme will operate only while nuclear stations are built under package contracts.

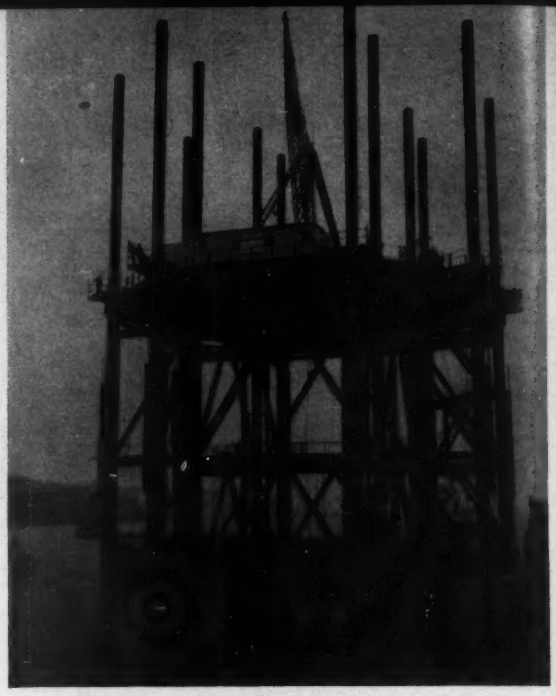
From August 15 onwards the use of sealed radiation sources in premises that are covered by the Factories Act, 1937, has to be reported to the local inspector of factories, as also does the discontinuance of their use. It is a requirement of the Ionizing Radiations (Sealed Sources) Regulations, 1961, which were presented to Parliament on August 3 and of which the other requirements are not enforceable until after January 31, 1962. Except for a slight re-arrangement of some of the paragraphs the Regulations are in effect the same as presented in the draft earlier this year and discussed in *Nuclear Engineering*, May (pp. 184-5). Copies of the Regulations (Statutory Instruments—1961 No. 1470) are available from the Stationery Office, price 9d.

Latest development in the university "reactor merry-go-round" (see last month's editorial comment, page 316) is that DSIR have decided to suspend for the time being the grants for the three proposed projects in London, Scotland and the north of England Also to suffer are two university radio-telescope projects. Although announced after the Chancellor of the Exchequer's "little Budget," the decision to suspend, it is understood, was made beforehand. The Department are apparently finding the demand for aid too great for them to meet all requests. They have said, however, that the university reactor grants will be reviewed in November.

A 4 GeV synchrotron to complement the work undertaken on Nimrod at Harwell is planned by the physics departments of the Liverpool and Manchester universities. The universities would operate it on behalf of NIRNS and it would be available to Glasgow University. A site in either Cheshire or South Lancashire is being sought and it is hoped to have the machine in operation by 1966. Estimated cost is £3 million. It is hoped to get Treasury approval for the project.

"Ministry of Procrastination" was a new name suggested for the Ministry of Transport in the House of Commons just before the summer recess after Mr. Marples had for the umpteenth time stated he was unable to make a statement on the Government's policy for nuclear marine propulsion. He did say, however, again not for the first time, that he would make a decision as quickly as possible. The matter had to be considered, he said, in the light of the Chancellor of the Exchequer's "little Budget" statement.

Negotiations should be concluded soon for the transfer of the Langley Jason to the Royal Naval College, Greenwich.



The 1 100 ton water intake structure for Dungeness nuclear power station being towed through Folkestone Harbour sound on its way to a site some 1 000 ft offshore from the station. There with its 12 legs lowered into the sea bed to a depth of 65 ft and reinforced with concrete, the structure will provide an inlet into a 13·5 ft dia. cooling water tunnel capable of taking 20 million gal/hr. The use of this particular type of structure instead of the conventional cofferdam was chosen on account of the risks involved from heavy seas in the English Channel.

A Code of Practice for the Display of Sources of Ionizing Radiations at Exhibitions has been prepared by the Radioactive Substances Standing Advisory Committee. Copies, price 6d. each, are available from the Stationery Office.

Loading of No. 1 reactor at Berkeley started on the morning of Saturday, August 12. Criticality was expected to be reached with 660 fuel channels loaded, 14 days later. There will then be a pause for three weeks while an evaluation of the reactor design will take place and tests carried out on control rods and the burst slug detection equipment. Loading will then be resumed in four stages until full load power is reached, probably early March. No. 2 reactor is about six months behind and loading is expected to start in December or January. Loading at Bradwell started on August 14. Criticality was expected to be achieved six days later with about 474 channels loaded. Full power operation is scheduled early 1962. Both stations are being built by The Nuclear Power Group.

Flooding in a road through the Berkshire village of Sutton Courtenay on August 1 was found to be caused through a rupture in a pipe used by the AERE, Harwell, for the discharge of waste into the River Thames. As the waste is sometimes active, although not on this occasion, the road was closed to traffic and health physicists called in to investigate. Some 5 cwt. of silt formed by the movement of the water through the soil (the pipe is about 6 ft below the surface) was found to contain something less than 1μc of activity, or in the words of one spokesman, "It would have been safe to drink one gallon of the most active of the sludge." The pipe, which is some 8 miles long and 18 in in diameter, was put out of operation and an AERE inquiry convened with Mr. A. S. White, head of chemical engineering at Harwell, in charge. At the same time the AEA emphasized that the incident was only of minor importance and no health hazard had been presented. Before the arrival of the health physicists children played in the water and cars splashed some of it on the walls of neighbouring houses, providing "scare" material for the more sensational newspapers.

Belgium

BelgoNucléaire's Vulcain reactor project has been submitted to Euratom by the Commissariat a l'Energie Atomique for advice concerning its inclusion in the Belgian national programme for the next five years. If the advice is favourable Euratom will be asked to co-operate in development of the project. The reactor has a potentiality in the marine propulsion field and has in fact been studied by the ENEA marine propulsion working group. On their recommendation BelgoNucléaire in conjunction with CEN are now working on an advanced design with a view to reducing capital costs. Although the Vulcain has been called a spectral shift reactor it is in fact different from the SSCR being studied by the American Babcock and Wilcox although both are mixed moderator PWR types. Whereas in the American unit control is effected through variations in the mixture of D_2O and H_2O, and by the use of control rods, in the Belgian design control is accomplished through temperature variations on the moderator. Shut down is effected through simple scram rods with a reactivity worth of 50% $\Delta k/k$.

Canada

Faith in the natural uranium, heavy water moderated and cooled reactor system for economic power production in Canada is reaffirmed by the Atomic Energy of Canada, Ltd., in their report for 1960-61. Referring to Candu, the report states that the AECL target of 9 750±10% MWd/tonne continues to be estimated burn-up attainable from this reactor. Other systems are not being ignored, however. The AECL is interested, for instance, in an organic cooled, heavy water moderated reactor and following an evaluation of a report by the Canadian General Electric on an OCDRE study together with information on organic systems from the U.S.AEC and Euratom the AECL will plan their course in this field. During 1960-61 AECL's expenditure amounted to about $42 million.

An explosion occurred in the chemical engineering building at Toronto University housing a low-energy reactor, on August 4, injuring two students. At the time of going to press the cause was unknown.

Installation of the strontium-90 thermo-electric weather station designed by the Martin Company for the U.S.AEC and U.S. Weather Bureau (*Nuclear Engineering*, August, 1961) in the Canadian Arctic was scheduled to take place during August. It was to be transported from the U.S. to Canada by truck and then taken to its site on Graham Island, near Norwegian Island, by the Canadian Department of Transport's ice-breaker, CMS *John A. MacDonald*.

West Germany

Amounts of DM16 million and DM18 million, respectively, have been guaranteed by the Bundes Ministry for Atomic Energy towards AEG's superheat BWR project and Babcock and Wilcox plans for a gas cooled reactor. Total cost of these projects are put at DM22·5 million and DM25 million, respectively. General Electric, U.S., are to participate in the design work on the BWR project.

Construction of the 50MW(e) Siemens multi-purpose reactor (MZFR) at Karlsruhe is to start shortly, according to Dr. S. Balke, Bundes Minister for Atomic Energy. The Ministry is to finance the nuclear portion of this project, estimated at about DM100 million, while the cost of the conventional electricity generating plant, about DM30 million, will be borne by several Baden-Württemberg utility companies.

India

Atomic Energy of Canada are to carry out with the Department of Atomic Energy a joint six-month study of the costs of a Candu reactor plant for India. Proposed site for the plant is in the north central region.

Italy

In return for having access to information on the 150MW Garagliano PWR plant being built for Senn by General Electric, U.S., Euratom is to make arrangements for the supply of fuel, the purchase of plutonium, chemical reprocessing and capital loans. The deal, concluded at the same time as the utility company signed a contract with the U.S.AEC, followed Euratom Council of Ministers' decision to allocate $7 million towards the project. The AEC contract provides for fuel guarantees over a 10-year period on the condition that Senn meets with certain requirements and requests for specific guarantees on individual cores. In return the AEC can have access to information. Euratom, incidentally, is following up its bargain with Senn by seconding five engineers to the plant purely for collecting data.

Japanese N-Ship Plan

A basic plan for the construction, financing and ownership of Japan's first nuclear-propelled ship has been drawn up. At the same time the Upper House of the Diet has prepared a Bill for the indemnification against damage by a nuclear ship. The construction plan, prepared by the Atomic Energy Ship Committee of the J.AEC, calls for the completion of a vessel by between 1968 and 1970. If the construction costs are to be borne by the Government then, the committee suggests, a budget should be asked for in 1963 and the basic design of the vessel completed within one or two years onwards. As to the ownership of the ship the committee makes three suggestions—a corporation with private capital, a specialized organization with Government investment, or joint ownership between Government and private enterprise. The committee, which held its first meeting in May, comprises representatives of shipbuilders, shipowners, electrical equipment manufacturers, bankers, atomic energy experimental groups, the universities and Government. The indemnity Bill provides for Government indemnification against damage amounting to over Y5 billion (about £5 million) to shipyard property caused by an incident in a nuclear ship. Provision is also made for Government compliance with any international indemnity agreements that are established.

Japan

Between £30 000 and £32 000 has been allocated by the Government sponsored Bureau of Science and Technology for two design studies on nuclear propelled shipping. One of the studies is of a 5 000 ton maritime research vessel equipped with a 35 MW PWR, and the other of a 45 000 ton, 20 000 s.h.p. oil tanker driven by a 65 MW BWR. The funds so far allocated are to last until March, 1962. In charge of the projects are engineers from the Mitsubishi and Hitachi Shipbuilding and Engineering Companies. In the meantime consideration is being given by the recently formed Atomic Ship Committee of the J.AEC to the 1962 budget for research into nuclear ship problems to be carried out by the Ministry of Transportation. It is estimated that something in the region of £175 000 will be needed.

U.S. nuclear submarines will not be allowed in Japanese ports under a recently announced Government order. The ban is generally regarded as being of a political nature rather than arising from any technical consideration.

Netherlands

The HFR at Petten and adjoining land have been transferred to Euratom to form the basis of part of the Community's Common Nuclear Research Centre. The transfer agreement also provides for the building of new research facilities on the site, which was previously owned by Reactor Centrum Nederland. The Petten Centre under Euratom control—the fourth, incidentally to be taken over—will be concerned mainly with general nuclear research although in the early stages work will be concentrated on the HFR which is due to go critical towards the end of this year.

Although Petten is partly to be taken over by Euratom the Reactor Centrum Nederland hope to be able to continue working on nationally inspired lines. This is made clear in the Centre's report for 1960 in which it is stated that national development is necessary if full value is to be given to the collaboration. RCN feel confident that the Dutch Government and Euratom share this view. A good proportion of the Centre's work is, in fact, already of an international nature. During the year a design for Orgel was prepared by the design and development department while in another section work was carried out on irradiation experiments for Dragon and two joint Dutch-Norwegian projects. Partnership with Norway is still strong.

Norway

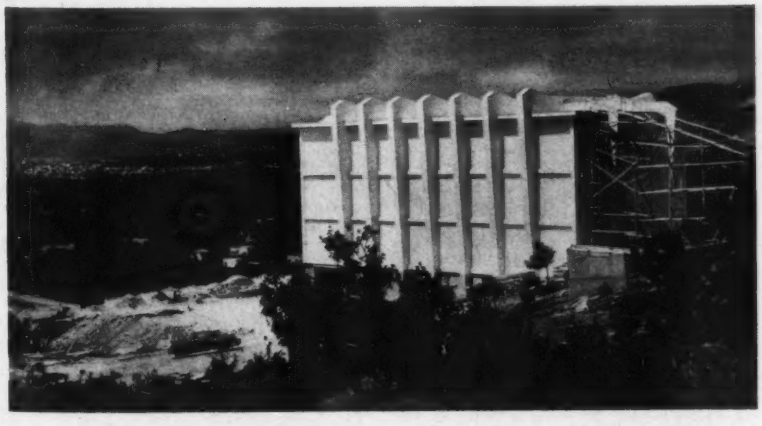
A cheque for $350 000 has been presented to the Government by the U.S. as a contribution towards the cost of Nora, the zero energy reactor at Kjeller. It is the 11th reactor grant to be made under the U.S. "Atoms for Peace" programme.

South Africa

The final order for the 5·5 MeV Van der Graaff accelerator for the Southern Universities Nuclear Institute has been signed with High Voltage Engineering, of Massachusetts. The cost is R330 000 (£165 000) and the accelerator is expected to begin operating in 18 months' time.

Sweden

Design studies for a 56MW(e) BWR, to be built at Simpevarp, south of Stockholm, being carried out by the Allis-Chalmers Manufacturing Company, U.S., for AKK. A decision on whether or not to go ahead with the project is expected in November.



The reactor building at the Democritus Nuclear Research Centre on the outskirts of Athens. It houses a 1MW AMF tank type reactor for which the U.S. Government have provided $350 000. The centre is named after the Greek philosopher who anticipated the concept of the atom 2 500 years ago.

Switzerland

While one company came into being another was wound up. On July 18 the Société Nationale pour le Développement des Applications Industrielles de l'Energie Atomique (SNA), formed by the three nuclear groups, Enusa, Suisatom and Thermatom, officially came into operation with stock capital of Sw.Fr. 3·2 million. About the same time it was announced that International General Electric (Switzerland) was being put into liquidation, the parent company IGE having decided it was serving no useful function. The Swiss subsidiary was formed in 1959 with the express purpose of playing a part in the development of nuclear energy in Switzerland and to this end participated in the Suisatom group which at the time had plans for the construction of a 27 MW BWR. In March, 1960, the Swiss Government suggested that the three groups then considering separate nuclear plants should collaborate on the construction of only one plant, and so came into being the Société Nationale and the decision to build a 30 MW(th) heavy water moderated pressure tubed plant at Lucens. General Electric's interest and collaboration thus came to an end.

U.S.S.R.

An annual output of 9-10 x 10^{11} kWh of electricity by 1970 and 27-30 x 10^{11} kWh by 1980 is called for in the Communist Party's draft programme to be presented to the 22nd Congress in October. Present output is somewhere in the region of 2·5-2·3 x 10^{11} kWh (in the U.S. about 9 x 10^{11} kWh; in the U.K. about 1·2 x 10^{11} kWh). Although no estimates are given as to how much nuclear power would play in achieving this target the Soviet programme states that as atomic energy becomes cheaper so construction of nuclear stations will be expanded, particularly in areas poor in other power sources. Under the programme, which is the third to be drawn up since the 2nd Congress of the Bolshevik Party in 1903, a single grid serving the whole of the Soviet Union and linked to grids in other socialist countries is envisaged. A research and development programme leading to such a grid is urged. Attention is also drawn to the need for developing methods of direct conversion of nuclear power to electricity and to solving the problems related to the control of thermonuclear reactions.

U.S.A.

Four paper producing companies have responded to the AEC's invitation for expression of interest in participating in a demonstration low temperature process steam project incorporating a nuclear reactor ("World Digest," July, 1961). Following an evaluation of information accompanying the response the Commission will decide whether to request for detailed proposals.

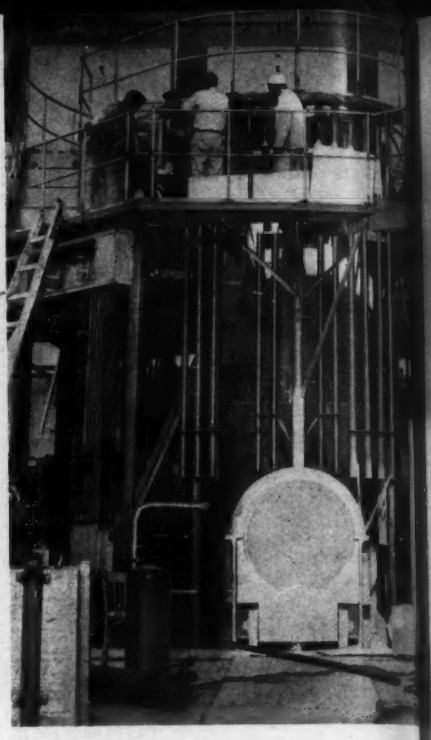
A plant cost analysis study of the spectral shift control reactor concept is to be made by Babcock and Wilcox under the terms of a $211 000 contract recently awarded to them by the AEC. Last year the company received a $1 223 500 contract for a basic physics study of the concept and now that it has been established that reactor control can be accomplished through using varying mixtures of heavy and light water it has been decided to go a stage further and investigate the economics of the system. By using the SSCR system Babcock and Wilcox have estimated that it will be possible to modify an ordinary 200 MW PWR to a 350 MW reactor and thus make a saving of about $100 kW or 2 mill/kWh for an 80% load factor. Fuel cycle costs of less than 2 mill/kWh, it is believed, could probably be obtained with low enrichment uranium fuel and it is further estimated that with a U-233 thorium fuel cycle the cost could be further reduced to about 1 mill/kWh.

Design work on a high flux research reactor for the National Bureau of Standards at Gaithersburg, Maryland, is nearing completion. The reactor is based on the CP5 and will have an initial output of 10 MW. With additional cooling towers and heat exchangers a power level of 20–25 MW will be feasible.

Congress is playing hot and cold over the authorization of $95 million for converting the Hanford Production Reactor into an 800 MW generating plant. The Bill containing the authorization made its first appearance on July 13 in the House of Representatives and was defeated by 176 votes to 140. A few days later the Senate supported it and at the same time added $5 million for a coal laboratory. The next step was to decide whether there should be a joint conference between the House and the Senate to iron out the differences and settle for a new Bill. By a 235 to 164 vote the House of Representatives on August 8 defeated the move to a joint conference, 81 Democrats joining in with the opposition. It is believed that the Democrats will now introduce yet another new and "substantially curtailed" Bill. This could mean either a cut or halving of the NPR requirements and/or a reduction in appropriations for other projects.

A portable self-regulating 2 MW reactor for generating electricity beneath the surface of the sea or in remote land areas is being designed and developed by the General Atomic Division of General Dynamics. Based on the Triga concept, it will use homogeneous fuel-moderator elements of uranium zirconium hydride.

A six-month $6·3 million contract for phase one of the joint AEG-NASA rocket nuclear project, Nerva (nuclear engine for rocket vehicle application), has been awarded to Aerojet-General. Under a subcontract Westinghouse Electric Corporation will be responsible for work on the nuclear portions of the project. The main contract calls for a preliminary design of the engine, performance of work in conjunction with the Los Alamos laboratory on reactor tests, assistance in the design of certain test facilities, preparation of a development plan, and research and development work on a pumping system, the nozzle, controls, bearings and seals and the safety aspects of the design. Concurrent with this work four other companies will be carrying out preliminary design studies on the NASA sponsored Rift project (reactor in flight test), $75 000 contracts for which having just been awarded to the Martin Company, Lockheed Aircraft, General Dynamics Astronautics and Douglas Aircraft.



The Soviet designed 2 MW tank type reactor at Inshas, near Cairo which recently achieved criticality. The reactor was built and put into operation by a joint Arab-Russian team.

A series of controlled fission product field release tests are to be conducted at the National Reactor Testing Station, Idaho Falls, during the next few months in order to improve predictability of the behaviour of fission products when released to the atmosphere at surface level. Instruments will be placed at various geometrically fixed points to measure the effect of diffusion conditions on downward air concentrations. No spread of radioactive materials is anticipated beyond the experimental area, which is situated in the approximate centre of the NRTS. The tests, which are part of the joint AEC-U.S. Air Force safety analysis programme, are to be managed by Convair, Division of General Dynamics.

What is known as the Black Void Reactor (BVR) concept has been the subject of a study by the National Bureau of Standards. Such a reactor would provide simultaneously both fast and slow neutron fluxes and so permit independent radiation effects studies and thermal neutron beam experiments to be carried out. In its simplest form the BVR would consist of a single cylindrical fuel assembly, about 8 ft long and with a diameter of 8 in, immersed in a large tank of moderating material. A fast neutron flux of about 10^{14}/cm^2 would be available in the middle of the cylinder for radiation effects studies while on the outside the neutrons would be slowed down by the moderator and enable beam tube experiments to be carried out. Before any practical application of the concept can be put in hand extensive development and testing of the annular fuel elements required will be necessary. Reactors that might be useful in such development work are the high flux beam reactor, under construction at Brookhaven, and the high flux isotope research reactor, construction of which has just started, at Oak Ridge.

AEC permission has been given to Kansas University for the installation of a Triga reactor between now and the end of 1962. It will be sited in the new nuclear science and engineering laboratories.

Norway, Greece and the Netherlands are the only countries with which negotiations for the acceptance of the *Savannah* in foreign ports have so far reached "an advanced stage," the AEC has reported to the JCAE. Apart from an agreement on staff level with Germany about the terms for liability provisions, little progress has been made with the rest of the 12 countries with which negotiations are being carried out, including Britain.

Reprocessing of heavy water for overseas reactor operators can now be undertaken by the AEC Savannah River plant on the condition that the heavy water was originally bought from the Commission. There are no commercial facilities in the States for this service.

Only one organization out of eight that had originally shown tentative interest in the AEC's proposals for a joint 50 MW OMR plant has maintained sufficient interest to respond to the Commission's invitation for definite plans. It is the Grand River Dam Authority, Oklahoma. If the plans are approved the Authority will provide the site for the plant, build the conventional generating equipment, operate the reactor and pay the AEC for steam consumed while the AEC will build and own the reactor.

Construction of the HFIR at Oak Ridge has started, it is expected to be ready for operation by 1964.

A new health and safety clause in reactor operation contracts has been proposed by the AEC. It would be more specific than the present one which calls for the contractor to "take all reasonable precautions . . . to protect the health and safety of employees and members of the public and to minimize the danger from all hazards to life and property." It also requires "compliance with all health, safety and fire regulations and requirements (including reporting requirements) of the Commission." Proposals for the new clause are at the present being discussed by industry.

A public hearing in connection with the issue of a provisional operating licence for the 5 MW(e) BWR plant at Saxton, Pennsylvania, has been called for September 6. Approval of the start-up programme has been given by the ACRS who, in their report, draw attention to the "possibility that reactor can be made critical by complete withdrawal of a single control rod." Since the SL-1 accident this has been considered a feature worth special scrutiny.

A modified Liberty ship is to be installed with a 10 MW(e) nuclear power plant to provide the U.S. Army with a floating power station for supplying electricity to military installations and towns cut off from normal supplies. A $16 998 127 contract to design and construct the plant has been awarded to the Martin Company, Baltimore.

Correspondence

Magnetic Flow Meter

Sir,

We note with interest Winfrith's development of a magnetic flow meter for non-magnetic fluids, slurries, dusts, etc. The article, on page 80, *Nuclear Engineering*, February, 1961, is not in detail and we are interested in more information.

Is this development at a stage where detailed information is available to us? Please advise.

D. W. MONTGOMERY.

Supt. *N.S. Savannah* Nuclear Technology, New York Shipbuilding Corporation, Camden 1, New Jersey. U.S.A.

Sir,

As you will be aware the principle of the conventional magnetic flow measuring instrument involves measurement of the e.m.f. induced in a conducting liquid when this liquid is made to flow normal to a magnetic field. The e.m.f. is detected by means of probes and requires that the liquid should have a finite conductivity. Although instruments of this type can be usefully employed even with liquids of quite low conductivity there is a range of liquids (for example, oils, dust suspensions and demineralized water) which may have such low conductivity that this type of instrument is unsuitable.

However, it may be shown that movement of an insulator normal to a magnetic field induces polarization of the material of the insulator. Thus, in a configuration similar to the conventional magnetic flow instrument, if the probes are replaced by plates so as to form a capacitor using an alternating magnetic field, an alternating e.m.f. will be developed across these plates. This e.m.f. will be developed in the finite impedance of the plate capacitance and may therefore be measured.

The capacitance between the plates is given by:

$$C = \varepsilon_o \varepsilon_r A/d$$

where ε_o is the permittivity of free space
ε_r is the relative permittivity of the dielectric
A is the plate area
d is the distance between the plates

This may be written as

$$C = \varepsilon_o A/d + \varepsilon_o(\varepsilon_r - 1)A/d$$

in which the first term is the capacitance which the assembly would have without the dielectric and the second term represents the capacitance contributed by the dielectric.

It may further be shown that the effect of polarization is equivalent to the appearance of an e.m.f. in series with the capacitance given by this second term. This e.m.f. is the same as would have been developed in a conductor; the e.m.f. developed across the plates is somewhat less than this by virtue of the first capacitance term, which shunts the system. For a given configuration this shunt capacitance is known and constant and is thus a constant of calibration. The measured e.m.f. in this case is proportional to the mean molecular velocity of the dielectric between the plates.

If the detecting device is designed to present a short-circuit to the plates (e.g. if a virtual earth input amplifier is used) the shunt effect of the first term in the above equation is eliminated (as is the effect of any other stray shunt capacitance). The current to the detector which then results is proportional to the product of mean molecular velocity and $(\varepsilon_r - 1)$. To the extent that $(\varepsilon_r - 1)$ is proportional to the mass of material between the plates then the measurement will be proportional to mass flow. This relationship tends to be true for dust suspension and also for certain liquids over a limited range of temperature.

Finally, it is worth noting that, if a partially conducting liquid is to be used, the unwanted effects of the conductivity current may be eliminated by phase-sensitive detection, as the capacitance and conductivity currents are in quadrature.

It should be pointed out that our device is at present in a very early stage of development. In fact, we have done no more than produce a simple model in order to check the basic principles. As we have no immediate requirement for an instrument of this type (although possible future requirements can be foreseen) further development at Winfrith cannot be justified at present. However, a number of industrial concerns in this country have expressed an interest and some of these are considering development action for particular applications.

D. HARRISON,
Control and Instrument Division,
AEE, Winfrith.

Another Reactor

Sir,

In your issue of July 1961 (page 272) you state that "It is understood that AEI are negotiating with Hawker Siddeley for the acquisition of the Jason 'know-how' and construction rights" and that "if a deal results it will probably mean that universities that have opted for this particular type of reactor will not have to go overseas for it."

May we draw your attention to the fact that our Company has recently announced that as a result of agreement between this Company and the American Machine and Foundry Company, U.S.A., a British-built AMF research or teaching reactor is available in this country.

We feel that your statement implies that there is no British alternative to Jason available to British universities. In fact, certain universities in this country have already expressed strong interest in the type of reactor which we are marketing.

V. B. HESSEN,
Sales Manager
(Engineering Division), Pye Ltd.

A Betatron Overlooked

Sir,

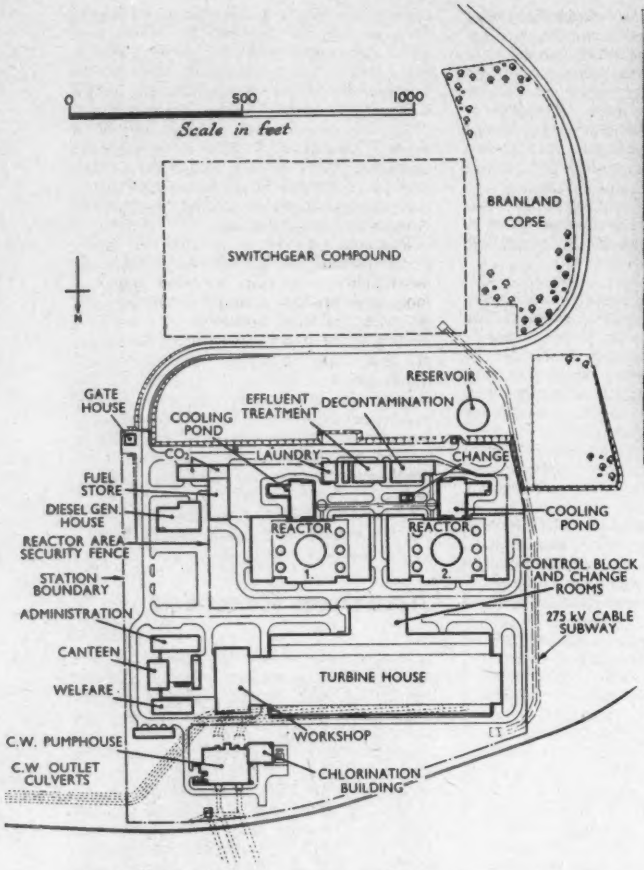
It was very interesting for me to read "Accelerators in Radiography" which appeared in *Nuclear Engineering*, November, 1960.

To my regret, however, no information about accelerators used in radiography other than those of western counties was included.

Already over 10 betatrons, almost all units of which are our products—15 MeV lightweight units—are successfully used in heavy industries in Japan. Our 15 MeV betatron was described in Toshiba Review, International Edition, January, 1960, a copy of which will be sent to any of your readers who are interested.

HIROSHI KAMOGAWA.

Atomic Energy Research Dept., Matsuda Research Laboratory, Tokyo Shibaura Electric Co., Ltd., 72 Horikawacho, Kawasaki, JAPAN.



View, looking east, of the 500 MW Hinkley Point station showing the two reactor buildings with the 740 ft long turbine hall on the left. (Left) Plan of the Hinkley site.

From Hinkley . . . to

HINKLEY POINT is the CEGB's third nuclear power station but the first of the group designed for a minimum output of 250 MW(e) per reactor. When construction was first begun late in 1957 the contractors, the English Electric-Babcock and Wilcox-Taylor Woodrow Atomic Power Group, hoped that the first reactor might be completed by the end of this year, but this made no allowance for any hold-ups nor could it be anticipated how serious would be the dislocation to constructional schedules caused by the AEA's changing views on graphite behaviour under irradiation. Altogether, the numerous revisions of the core design have put back the project by more than a year and fuel loading of number one reactor is now planned to begin in February, 1963, with full power operation by the following October. Number two reactor is programmed to follow six months behind, although on the basis of present progress the gap could be narrower.

Apart from the core, the number of significant design changes made to Hinkley since the contract was placed has been small. Some alteration to the instrumentation was necessary once the implications of radial instability were appreciated but the decision to install three additional ion chambers in vertical channels in number one reactor to give the necessary 120° flux measurement, was taken in time for the alterations to be made without too much interruption to the pouring of the biological shield. It was at an early stage also that the CEGB decided that reactor No. 2 should be fitted with a great deal more instrumentation than would be necessary simply for operational purposes; extensive studies are to be made of the changes in the pressure vessel shape over its life, movements of the graphite core and so on, and some 60% of the total instrumentation on reactor No. 2 is to be for research purposes. Considerable use is being made of resonance capacity strain gauges within the reactor vault, a policy open to criticism on the grounds that long-term experience with them is limited and no maintenance is possible, whereas a simple mechanical or optical lever is virtually foolproof.

At one time it looked as if the Government's plan to provide for the extraction of military grade plutonium would affect the design of the pile cap area and refuelling machinery, but the order was rescinded before much work had been done. Other modifications to the plant of minor importance concern the standby generators, increased from three to five and diesel driven; additional containment for the ducting to minimize the hazards in the event of a duct fracture and also to improve access to the boiler areas.

On the whole the 12 months separating Bradwell and Berkeley from Hinkley have been to the advantage of Hinkley, as final decisions regarding the core could be left until the graphite data had become more complete. This is not to say that the contractors were not required to go through the sleeve-no sleeve exercises and the growth-shrinkage *volte face* and the final core design is still something of a compromise. Nevertheless, costly and annoying though the delays must have been, at least it meant that the building programme could more easily be modified to take into account site hold-ups and also to evaluate thoroughly some of the materials and constructional problems that arose instead of seeking emergency cures without proper time for consideration. The second reactor is already reaping the benefits of the experience gained on No. 1, as also is Trawsfynydd, for which Babcock and Wilcox are the pressure vessel contractors.

Pressure Vessel Erection

The largest and most critical item of a gas cooled graphite moderated reactor is the pressure vessel. The Hinkley vessels are spherical, 67 ft o.d., built up of plates of maximum thickness 3 in. The top and bottom domes and the five horizontal courses were prefabricated in a site workshop and transported into the biological shield by a Goliath crane with a span of 250 ft and a capacity of 400 ton. The plates were first pressed to shape at Renfrew, latterly welded in threes and shipped by sea to Combwich, from whence they were transported to site by

road. Delivery of these plates was behind schedule, as also were some pressure vessel forgings, because of a last-minute change in impact test requirements. Plates are in a silicon killed 0·2% C, 1% Mn steel in which ½ lb/ton Al has been added to the ladle; forgings are of a similar steel except for aluminium additions of 1-2 lb/ton. The AEA would have preferred a different analysis and a greater degree of grain refinement to improve notch ductility but in the light of their previous experience the contractors preferred a steel which they could expect to be cleaner and less prone to laminar-type defects and hydrogen cracking.

Original schedules called for a number of simultaneous operations inside the pressure vessel but experience has shown that it is unwise to conduct more than one major activity at any one time within the shell. Surveying the shell is an exacting task but the system finally adopted has proved highly satisfactory in practice. The template which had been prepared for covering the diagrid was abandoned as it was found difficult to install it in an undistorted form. Jig-bored plates were used to survey the diagrid itself and this survey was then transferred to the top dome optically, following which the rough burnings out were made and the nozzle apertures accurately bored. The standpipe stubs were accurately positioned on the nozzles for welding by means of a damped plumb bob suspended down through the nozzles to the diagrid markings. This survey was

. . to Sizewell

then transferred on to the Callender-Hamilton bridge which had been constructed over the reactor vault for holding the top thermal shield and the permanent shuttering for the top biological shield.

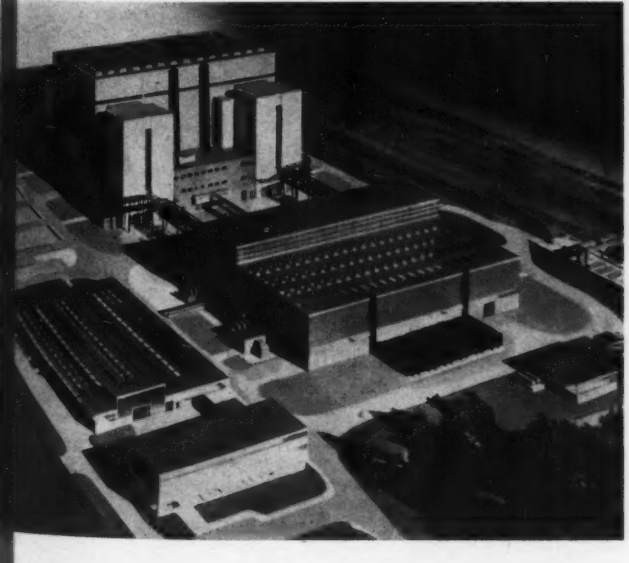
Very tight specifications had been set on the location of the tops of the standpipes and it was found in practice that the small distortions of the pressure vessel and the tilts introduced when welding, magnified over the length of the standpipes, caused movements significantly exceeding the design tolerances. For a time shimming of the lower surfaces of the standpipes was not permitted and it was found necessary to re-evaluate the permitted deviations from the ideal and to introduce a more careful system of jacking and balanced welding. The second evaluation showed that the original design tolerances had been excessively tight and the relaxation of these, in conjunction with the revised welding techniques, permitted the new design criteria to be met. In setting the nozzles some allowance had been made for expansion and take-up during construction, and with the experience now gained on Hinkley this allowance can be made on future vessels with a much greater degree of accuracy than was possible by estimation alone.

Boiler Construction

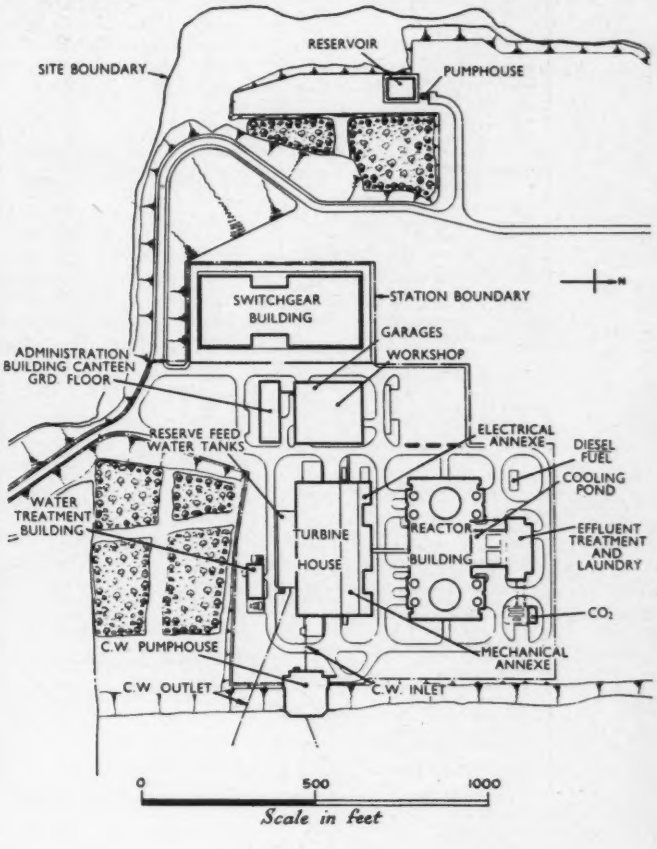
On the boilers also, experience has shown the way to simplifying and speeding the constructional programme. The first boilers were cleaned in the horizontal position and difficulty was found in removing continuously the scale and dust. By leaving the cleaning operation until they had been swung into their vertical position the cleaning process was much simplified; it is largely as the result of this that the last six (on No. 2 vessel) were put into position, after completion of the welding, in half the time that was required for the first six.

The shells of the boilers are, of course, pressure vessels in their own right and stress relieving is an important part of the manufacturing process. It was felt that particular regard should be paid not just to the maximum temperatures attained but also to the temperature gradients within the vessels. A great deal of attention was paid, therefore, to the method of heating and positioning of the insulation blankets to get these temperature gradients right. Stress relieving of the first boiler took several weeks but once the scheme had been properly devised it was found possible to stress relieve each boiler in less than ten days.

Like Calder, the extended surface tubing in the heat



A model of the 580 MW nuclear power station at Sizewell, Suffolk, where both reactors are in the same building. (Right) Plan of the Sizewell site.



exchangers is made by an automatic stud welding process. The studs for Hinkley are of " lozenge " cross-section, 1 in high and 1 in across. As the stud has a straight edge and is welded to the 2 in tube, there are crevices at the ends of the welds and it was found on shot blasting the tube banks that there was a tendency for shot to get trapped there. This problem was circumvented by changing to acid cleaning at site, and now that such niceties of working have been introduced as providing terylene ropes for handling the jigs carrying the tube banks, the operation is streamlined to a simple routine.

Main Ducting

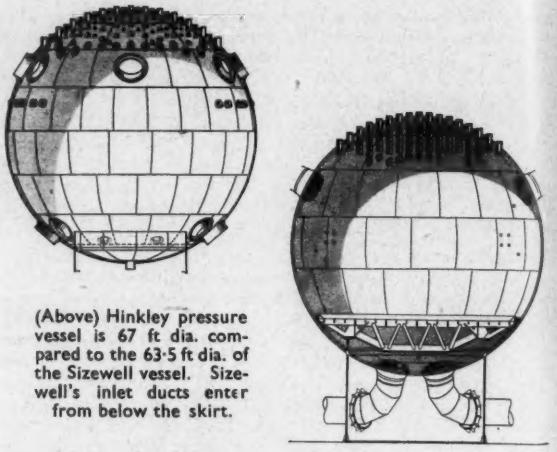
During the early stages of welding on the large ducts carrying the primary coolant, difficulty was experienced in meeting radiographic requirements. This was mainly due to the radiography being carried out to a sensitivity of 0·166% as opposed to the normal standard of 2%. The effect of this was to show up the minutest defects and, as the acceptable standard was nothing less than a clear film, this led to a cut-out rate above the average.

Another difficulty experienced was due to the weld preparation in the ducts. Several methods were experimented with, and it was found that a single vee-weld preparation, as opposed to the double vee, produced better results. Further to this modification in design, an intensive check on the results of each welder on this particular class of work was made, and a team was built up which resulted in a more consistently high standard of welding being maintained.

STATION PARAMETERS

Property, unit	Hinkley Point	Sizewell
Guaranteed output, MW(e)	500	580
*Thermal output MW(th)	980	950
Net thermal efficiency, %	25·5	30·5
Mean fuel rating, MW/t	2·55	2·95
Peak fuel rating, MW/t	4·68	4·74
Peak heat flux, kW/l	14·8	15·3
Max. can surface temp., °C	430	454
Core dimensions, ft	49 dia.×25	45 dia.×26
*No. of fuel channels	4 500	3 800
Lattice pitch, in	Square, 7·75	Square, 7·75
Channel diameter, in	3·85	3·85
Fuel rod dia. in	1·10	1·10
Fuel rod length, in	36	41
Cladding	A 12	A 12
No. rods/channel	8	7
*Fuel change, t	376	321
*Wt. of graphite, (total) t	2 475	2 237
*No. of safety rods	16	8
*No. of control rods	129	99
*No. of control zones	12	9
Control rod material	Steel, or steel with B steel inserts	Steel, or steel with B steel inserts
Pressure vessel dia., ft	67	63·5
Pressure vessel steel	28/32 Si killed boiler	JTA 101 (28/32)
Pressure vessel plate, in	3	4¼
Pressure vessel max. temp., °C	344	344
Coolant pressure, psia	200	279
Coolant system pressure drop, psi	8·63	11·09
*Gas mass flow, lb/s	10 300	9 854
*No. of coolant loops	6	4
Reactor inlet temp., °C	180	214
Reactor outlet temp., °C	375	410
†Boiler dimensions, ft	21·5 dia×91	22·5 dia.×91·5
Shell thickness, in	1¼–2¼	2¼
Plate	28/32 Si-killed boiler	L.a., Mn, Cr, Mo,V electric furnace
†Heat transferred, 10⁶Btu/h	560	824
Tubing, in	2 i.d. studded	2 i.d. studded
Stud height, in	1·0	0·75
Stud section (overall), in	1×3/32	1×3/32
Stud pitch, /ft	288	288
Tube pitch, in	5	4¼
Circulator type	Single stage axial	Single stage axial with variable stagger inlet guide vanes
Drive	v.f. motors	squirrel cage induction motors
Power per blower, h.p.	7 000	8 700
H.P. steam pressure, psig	640	706
H.P. steam temp., °C	364	391
†H.P. steam pressure flow, lb/h	293 000	540 000
L.P. steam pressure, psig	171	254
L.P. steam temp., °C	350	389
†L.P. steam flow, lb/h	137 000	190 000
Feed water temp., °C	72	120
Installed capacity, MW	6×93·5+3×33	2×324·75
Vacuum, inHg	29	29
Thermal shield top, in	6, concrete	½, steel
Thermal shield sides, in	6, concrete	½, steel
Biological shield top, ft concrete	11	11·75
Biological shield main sides, ft	6	5·5
Biological shield supplementary sides, ft	5	5
Boiler shield, ft concrete	—	1·5

* Per Reactor. † Per Boiler.



(Above) Hinkley pressure vessel is 67 ft dia. compared to the 63·5 ft dia. of the Sizewell vessel. Sizewell's inlet ducts enter from below the skirt.

Comparative Costs

These then, are the sort of problems that have been met and resolved during the course of construction at Hinkley and it was against this background of experience that the tenders for Sizewell were prepared. Cost estimations for Hinkley were bound to be somewhat speculative and on the site construction side there was a considerable underestimation of the work involved. The true savings represented by the Sizewell design are therefore more impressive than they appear at first sight. The contract price for Hinkley and Sizewell it is believed were roughly the same, viz. £55 million, but Sir Christopher Hinton has quoted the overall cost of Hinkley (including site development charges, escalations, etc.) as £67 million. The output of Sizewell (580 MW) will be 16% more than Hinkley and the quotation includes site labour charges on the heavy erection work of something like double those calculated for Hinkley; there has too been a general increase in the level of wages of about 10%. Furthermore, at the time the Hinkley tender was prepared the British programme called for 5 000 to 6 000 MW by 1965. Now this has been revised a much higher proportion of the research and development charges must be added to any single contract—development and research work at Whetstone alone is reputed to cost £2 million a year. In addition to this the Group was prepared to construct Hinkley virtually for cost, but this state of affairs could not continue indefinitely and the Sizewell contract includes a normal margin for profits. Overall then, the estimate for the capital cost of Sizewell per kilowatt installed is some 30% less than the expected cost of Hinkley.

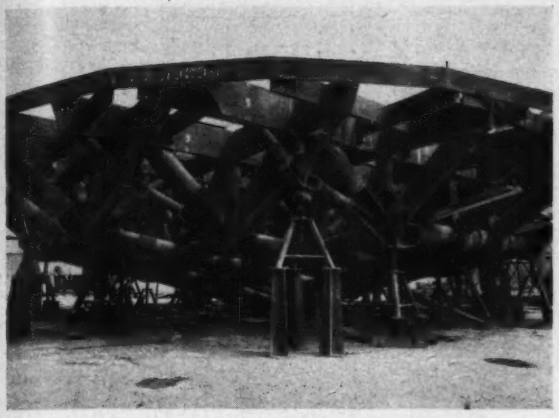
DESIGN EVOLUTION

The most important contributions to the rather dramatic cost reduction are:

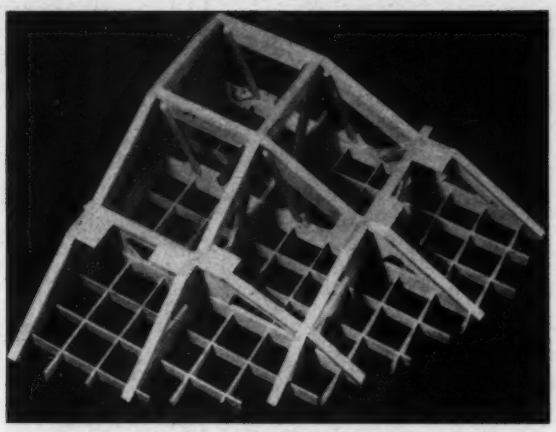
1. An increase in the station output brought about partly by
2. an increase in the coolant gas pressure allowing
3. a reduction in the number of primary loops from six to four;
4. the combination of two reactors into one major building with a common charge floor;
5. positioning of the circulators within the boilers which allows a simple main duct layout;
6. the use of two near standard turbines in place of the six sets at Hinkley;
7. attention to design details;
8. know how.

Station Layout

The latest regulations in the U.K. require that radiation levels outside the reactor building be not more than 0·25 mrem/h, the same level below which it is unnecessary to control the movement of personnel. The logical development of this specification is to eliminate the safety fence around the reactor area if the control of personnel moving in and out of the reactor



Welded diagrid supporting the graphite core at Hinkley.



A model of the Sizewell diagrid (shown here inverted).

area itself can be effected by some other arrangement. In the design for Sizewell such a layout has been developed, with both reactors under one roof and the charge room and control points located at the entrance to the building. Design studies on the plant and gas circuit for Sizewell indicated that it would be possible to obtain the required output from each reactor, equivalent to 290 MW(e), using four boilers and gas circulators. In addition, a gas circuit layout was developed which eliminated the two blower houses associated with separate gas circulators and horizontal shafts.

It was immediately appreciated that this compact arrangement could result in a reactor system layout in which the overall dimension across the centre of the reactor from the outside wall of the boiler hall on the one side to the outside wall on the other was less than 250 ft. This was significant as a Goliath crane is considered essential for the construction of reactor pressure vessels and boilers to achieve the shortest possible construction period, and this dimension could be readily spanned by the existing Goliath crane used at Hinkley Point. Moreover, by turning the reactors, as located in earlier designs, through 90°, it became possible either to develop a single, common irradiated fuel discharge and storage area between the reactors, or to accommodate a single common equipment building between the reactors. Further co-ordinated work by plant designers and civil engineers resulted in a closely integrated layout which maintained both a single irradiated fuel storage pond and a common equipment building.

The grouping of the facilities which together form the reactor area are dealt with differently at Hinkley Point and Sizewell. At Hinkley, the reactor buildings are quite separate. Each building contains an equipment annexe, normal and emergency fuel element discharge shafts, a maintenance shop, and is served by a separate cooling pond. A safety fence is placed around the two reactors, and along the line of the fence the charge rooms, fuel store, laundry and CO_2 stores are located. Within the fence, but separated from the reactors, are the effluent treatment plant, decontamination facility and stores for solid active waste disposal. The whole reactor area occupies about 9 acres.

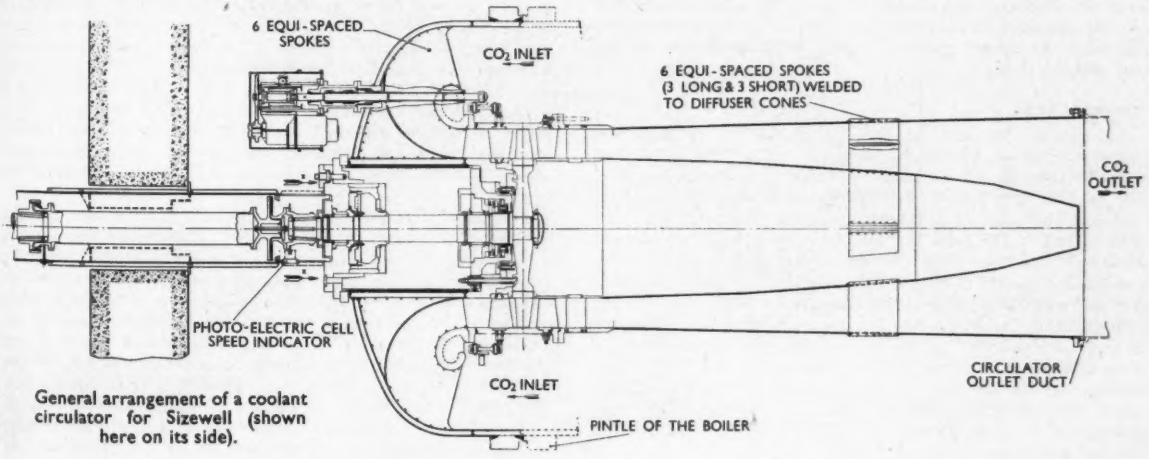
The same facilities are provided at Sizewell within an area of 5 acres. The two reactors are linked by a common equipment building and placed under one roof. Within the combined building, space has been planned for the fuel store, and the maintenance and decontamination areas. By centralizing the normal and emergency discharge shafts, a single cooling pond common to both reactors is used. The laundry, active solid waste disposal and liquid treatment plant are situated next to the pond to maintain compactness. The only separated ancillary building is the CO_2 storage plant house.

A further improvement at Sizewell is the linking of the station central control room with the reactors rather than the turbine house as at Hinkley, and the inclusion within the reactor building of the essential electrical supply for reactor safety—that is, the diesel generator and battery system.

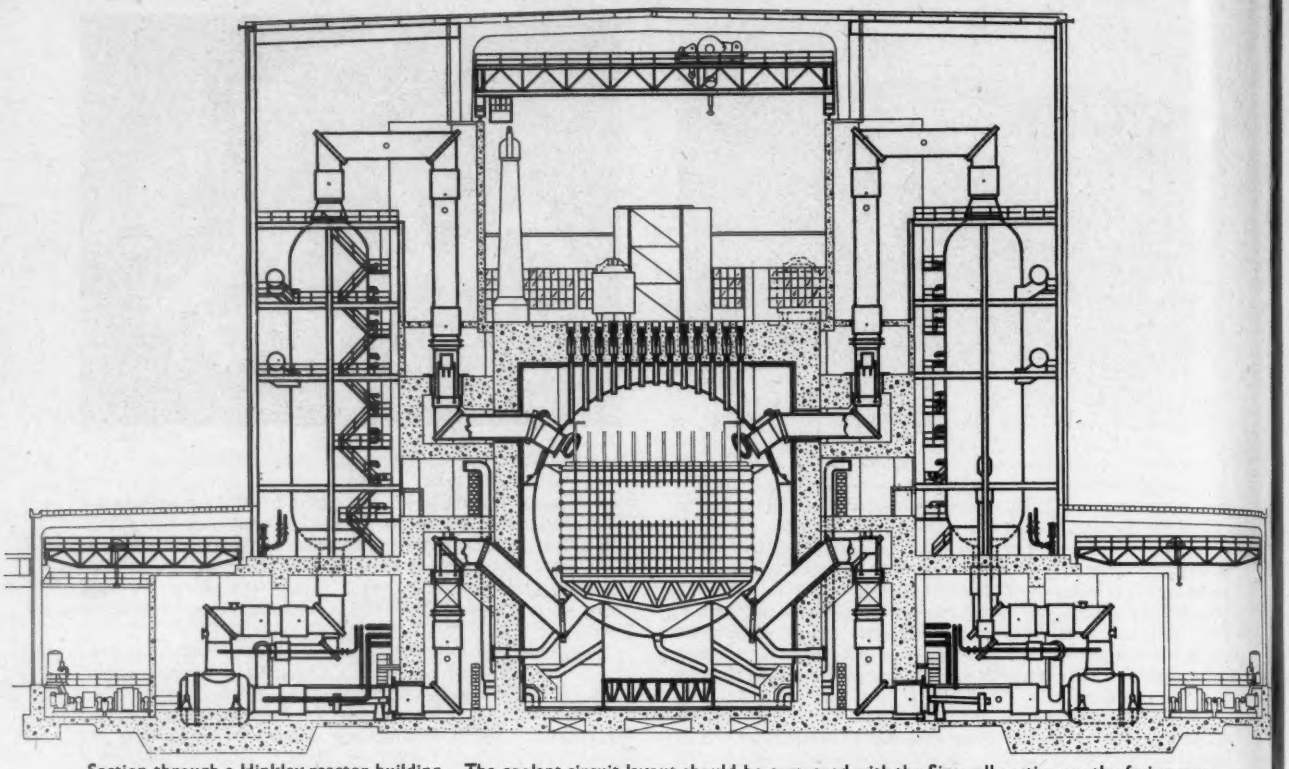
Additional simplification results from the installation of two 324·75 MW capacity turbo generators instead of the six sets at Hinkley.

Reactor Core

At Sizewell the primary coolant CO_2 will circulate at a pressure of 280 psia (as against 200 psia) through 3 800 fuel channels in a core 45 ft diameter by 26 ft high. Fuel elements may be of the stacked polyzonal spiral finned type with straight splitters as at Hinkley only longer (41 in as against 36 in and


6 EQUI-SPACED SPOKES
CO_2 INLET
6 EQUI-SPACED SPOKES (3 LONG & 3 SHORT) WELDED TO DIFFUSER CONES
CO_2 OUTLET
PHOTO-ELECTRIC CELL SPEED INDICATOR
CO_2 INLET
CIRCULATOR OUTLET DUCT
PINTLE OF THE BOILER

General arrangement of a coolant circulator for Sizewell (shown here on its side).



Section through a Hinkley reactor building. The coolant circuit layout should be compared with the Sizewell section on the facing page.

seven instead of eight to a channel), but considerable attention is being given to the herring-bone type becoming more popular in France. It seems quite possible that this type of finning will have developed to the point where it could be used for the first charge for the station. Its heat transfer properties are not very different from the polyzonal element but the great advantage is that, apart from minor effects due to small asymmetries in manufacture, there is no net radial force or moment applied to the fuel element.

The graphite core of Hinkley is of the block and double tile construction as used at Calder, but now that it is appreciated that shrinkage rather than growth will occur, each tile is slotted to accommodate two zirconium pins which traverse nearly the full width of the tile and abut against the side of the adjacent tile. To all intents and purposes the lattice dimensions are maintained by these zirconium pins as the two pairs of pins in the two tiles are at 90°. At Sizewell there are no tiles and the stack is made up of octagonal blocks with interlocking keys and keyways on the vertical faces. This is the generally adopted modern approach and as elsewhere the use of rolling supports has been abandoned, the stack being built up directly on the core support plates.

Pressure Circuit

To contain the core at the higher coolant pressure the pressure vessel will again be spherical but smaller—63 ft 6 in diameter fabricated from plate 4½ in thick. Steel for the plate will be JTA 101, a 0·1% C 1·4% Mn steel which is not grain refined. The steel is not very different therefore from the Hinkley steel. The same principle of support has been adopted whereby a cruciform forging, butt welded to the bottom dome and the first course of plates, transmits the load from the inner skirt carrying the diagrid to the lower skirt, both of which are butt welded to the cruciform. Temperature gradients within the forging are controlled by bolting to gunmetal studs three ½ in aluminium conduction plates inside and outside the lower skirt; thermal stresses are thus mainly confined to bending stresses within the skirt itself. To test and prove this method of construction a full-scale model was built at Renfrew before the Hinkley design was finalized.

An important departure from the Hinkley practice is the positioning of the inlet ducts. With six ducts it was necessary to make the inlets outside the support skirts but when the number was reduced to four it had been found practical to bring these through the skirt into the bottom dome.

The diagrid supporting the core is of generally the same construction—an open bridge structure—as at Hinkley but the new design is simpler, avoids complicated welding and is also lighter. The Hinkley diagrid was originally designed with three main trusses (per half section) but the U.K.AEA requirements for deflection limitation made a fourth truss necessary. It has now been agreed that the stringency of the limitations on deflection was exaggerated and Sizewell will have three trusses per half. The general construction is of tubular members through webs with a much simplified junction piece where truss meets truss.

Plates for the pressure vessel will be of the same size as Hinkley (a function of the press capacity) but it may not be possible to weld three together at works as done for Hinkley because of problems of road transport. At Hinkley the charge nozzles were made from forged billets but at Sizewell they will be made from rolled plate electro-slag welded. The main duct nozzles will also be fabricated, from bent forged bar electro-slag welded. Internal insulation of the vessel will be similar, viz. layers of stainless steel with expanded metal in between, but an improved method of fixing has been devised which should save much valuable time at a critical stage of the construction process.

The layout of the primary coolant ducts is much cleaner than Hinkley and is made possible by the reduction of the number to four and the enclosure of the main circulators within the boiler pressure shells. The same type of pin-jointed expansion units are included but apart from the lowest bend the cascade corners as used at Hinkley are replaced by lobster back bends. These are, if anything, easier to fabricate and like the ring reinforced cascade corner devised for Hinkley do not involve the welding together of sections of pipe of different thicknesses. The corner is included at the bottom to simplify débris removal.

The additional duct containment provided lately at Hinkley is already included in the Sizewell design and to a greater

extent. The official policy behind this duct containment is to limit the access to the primary circuit system to a few places which can readily be stopped manually in the event of a duct failure, so that fission products can be entirely contained in the building. The severity of accident which must be contained seems continually to increase and the present requirement that the design should provide for the appearance of a hole in a duct equal to twice its cross-section is bordering on the absurd.

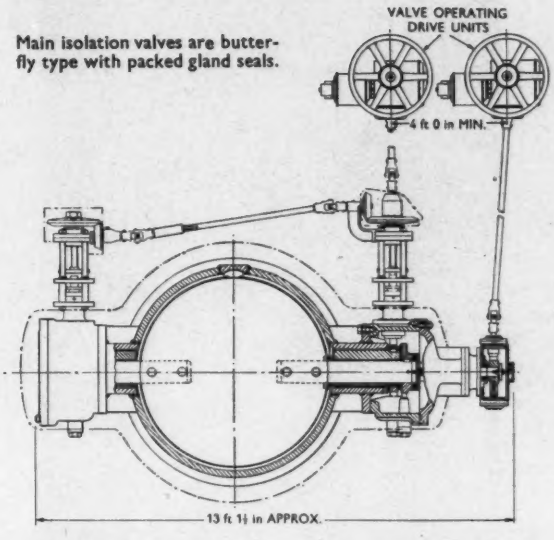
The main isolation valves are butterfly with packed glands as at Hinkley, the general policy adopted towards the gland sealing being that those in the ancillary circuits which are seldom used are capped (manually) whilst those which are relatively frequently used are bellows sealed.

There is a major change in the system of flow control. At Hinkley the blowers are driven by variable frequency motors supplied from auxiliary turbo alternator sets; at Sizewell the circulators will be driven by constant-speed motors, the flow being controlled by a by-pass valve inside the boilers. There is an external by-pass circuit at Hinkley but this is for start-up.

Steam Raising

The boilers at Sizewell while not substantially larger (only 1 ft in diameter) will have a heat transfer capacity more than 50% greater than those at Hinkley. This has been made possible by (1) the increased pressure of the coolant, (2) an improvement in the heat transfer properties of the tubes in terms of stud configuration leading to (3) a tighter arrangement of the tubes in the banks made possible also by (4) a relaxation of the limits set on maximum pressure drop. Pressure drop through the Hinkley boilers is less than 1·5 psi whereas at Sizewell it will be approximately 2·5 psi. The stud height for the Sizewell tubes has been reduced from 1 in to ¾ in and while the studding pitch remains at 288 studs/ft the pitch of the tubes has been brought down from 5 in to 4¼ in. The heat transfer per foot length is, in fact, 20% lower than at Hinkley but the overall gain is considerable. In the Calder boilers where the banks were

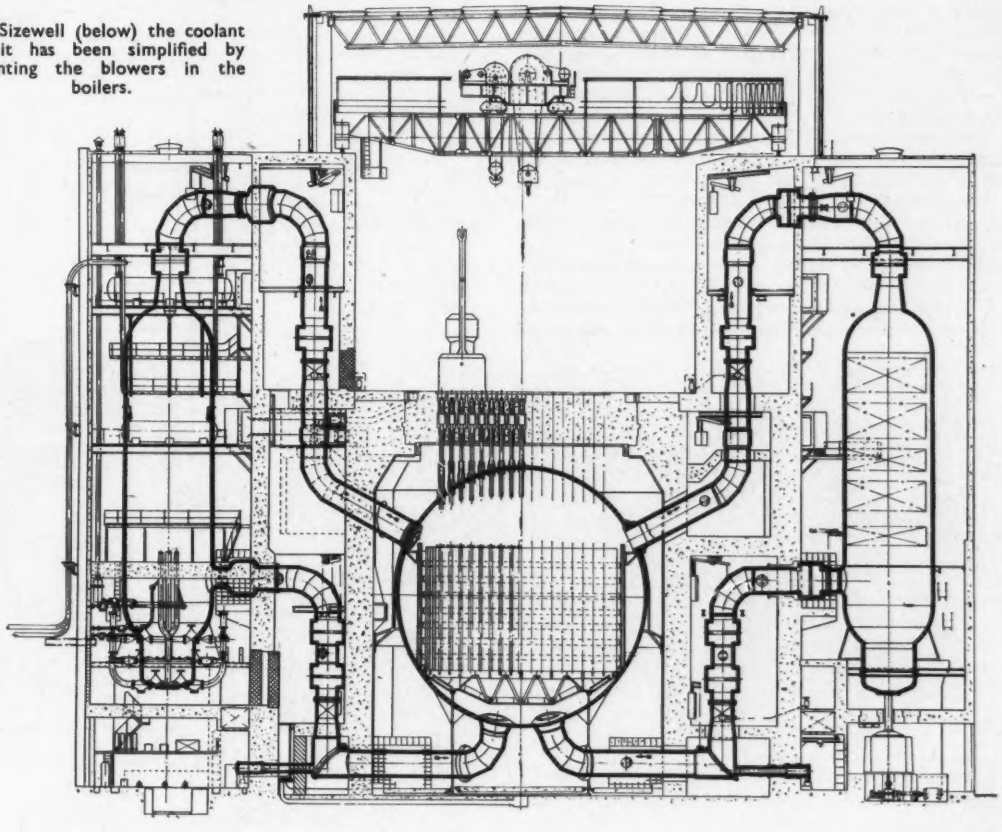
Main isolation valves are butterfly type with packed gland seals.



contained within a central square duct only 40% of the total area was used, but this was increased to 80% at Hinkley by eliminating the duct and installing tube banks of different lengths. At Sizewell, by changing slightly the angles of the thermal sleeves the packing problem has been simplified and 85% of the area will be utilized; moreover, the elements are trifurcated instead of single so the number of entries and exits has been much reduced.

The overall dimensions of the pressure shell are 22 ft 6 in diameter by 91 ft 6 in high. At the operating pressure of

For Sizewell (below) the coolant circuit has been simplified by mounting the blowers in the boilers.



Hinkley's Goliath—with a span of 250 ft and a maximum lift of 400 ton—will also be used at Sizewell.

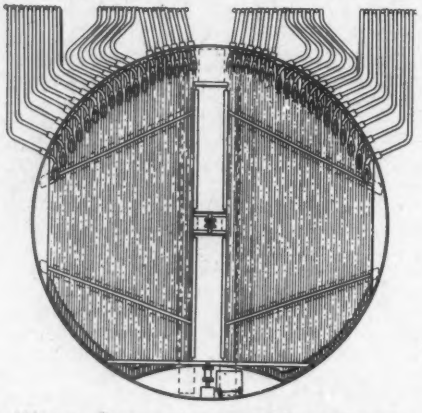
280 psia the difference in price between a mild steel and a low alloy steel vessel is small. During construction, however, the same Goliath as used at Hinkley will be employed for lifting the boilers into position and a mild steel construction would have involved a weight over the limit of 400 ton. In the event, a Mn, Cr, Mo, V electric furnace steel with a minimum UTS of 34 ton/in^2 will be employed. A steel similar to this has been used in high pressure boiler drums for many years in thicknesses up to 5$\frac{8}{8}$ in so that welding the 2$\frac{1}{4}$-in plate in the boilers should present few problems. It is worth noting that the first Sizewell boiler will be the 45th for Calder type reactors to be made by Babcock and Wilcox.

Substantially the steam cycle in the two stations will be the same but the disposition of the tubing inside the boilers is slightly different in that at Hinkley the superheater sections are in series whereas at Sizewell they will be side by side; forced circulation is used in both. An interesting provision inside the boilers additional to the by-pass duct for flow control is a second internal by-pass. The object of this is to raise the inlet temperature to the reactor not, as it might at first be supposed, to initiate a Wigner release (should this unexpectedly prove necessary) but to take advantage of the positive temperature coefficient on high burn-up, if it should prove difficult to obtain xenon override after a shutdown.

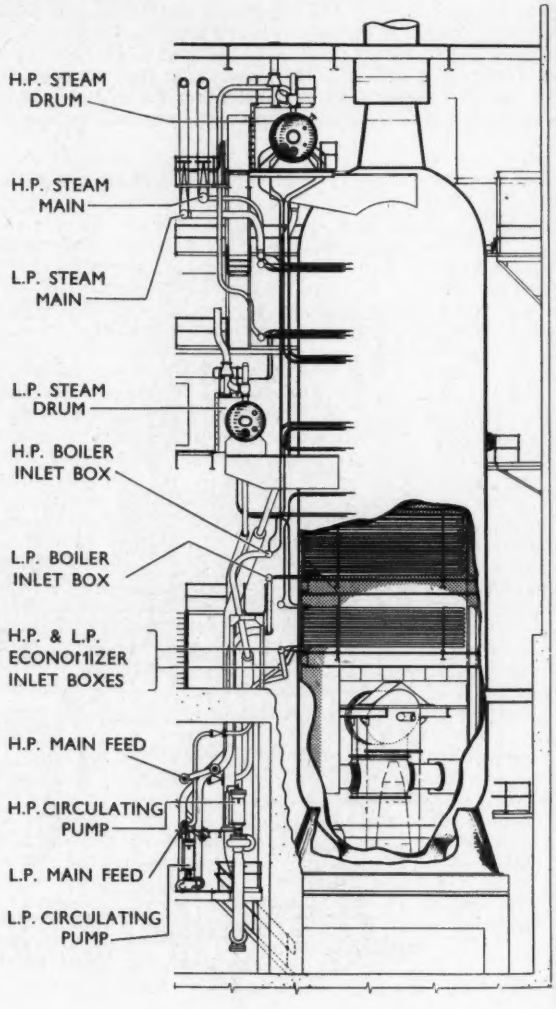
The Sizewell circulators are vertical single-stage axial units with variable stagger inlet guide vanes mounted within the boilers and driven by 8 700 h.p. squirrel cage induction motors supplied from 11 kV unit transformers. The circulator main shaft face seal is of the more conventional oil-backed type. At Hinkley the Generating Board were keen to see tried out a gas seal but apparently have not continued to encourage its use at Sizewell.

Civil Engineering

On the civil engineering side, Taylor Woodrow have been able to call upon their experience of the largest conventional stations and Calder Hall. The general construction of Hinkley was supplemented by such work as the provision, at an early and vital stage, in heavy tidal conditions, of a 3 500 ft sea wall. Later an intake structure 2 000 ft off-shore was erected to serve the needs of this and any future stations on the site. Techniques similar to those employed at Hinkley, refined where appropriate, are being applied on the more compact site at Sizewell, where a sea wall is not required. One interesting point that should be mentioned is that the vertical sea-water cooling shafts which join up with the horizontal tunnels will be driven upwards instead of downwards as was the practice but which is now being steadily outmoded by the new method.



(Above) Cross-section through Sizewell superheater. (Right) Side elevation of the Sizewell boiler.



An Economic Fast Reactor

By Dr. L. R. BLAKE*
(*Director of Engineering, Brush Electrical Engineering Co. Ltd.*)

With only a small extrapolation of present-day knowledge the author of this article shows how a fast reactor could be built to produce power at about 0·32 d/kWh. The crux of the design is the use of a voided fuel (in the first instance uranium-plutonium metal alloy) enclosed in a strong can the practicality of which has already been demonstrated.

OVER-ELABORATE design has made existing fast reactors more expensive than necessary, but even at the first attempt, the capital cost achieved in the Enrico Fermi reactor was $59 million, including an R and D surcharge. Based on the expected ultimate output of 150 MW(e) this is about $392/kW installed, or about £100/kW under U.K. conditions. Table 1 shows the capital cost breakdown of the Enrico Fermi Reactor[1], together with a capital cost estimate for a 250 kW(e) plant designed by the author. This suggests that it should be possible now to achieve in the larger plant about $240/kW (or £60/kW) and this is confirmed by the Fermi designers[1]. Substantial savings over Enrico Fermi are made possible by the larger size (this alone should reduce Fermi to $320/kW), simplifications in fuel handling, storage and in the reactor vessel, the use of electromagnetic pumps, and the provision of fewer auxiliary facilities.

At £60/kW, the capital cost of the fast reactor compares favourably with that of the lowest capital cost thermal reactors, such as the pressurized and boiling water reactors and even the organic moderated reactor[2]. This is to be expected as the low capital cost of the fast reactor is due mainly to its inherently compact core and its use of sodium as a cooling medium, which makes for efficient heat-transfer and keeps the system pressure substantially atmospheric. The figure of £60/kW is not the lowest limit; experience should enable £50/kW to be achieved without much difficulty.

Fig. 1 shows now the variation of fuel cost and its components with burn-up. This estimate is based on a 750 MW(th) metal-fuelled fast reactor operating at 111 W/g and 700°C, the important design parameters of which are given in Table 4, column h. The main assumptions are:—fuel costs: £6/g Pu^{239}; £10/kg depleted U; reprocessing and refabrication costs: £200/kg fuel; £20/kg blanket; 150 days from unloading to reloading core fuel; 0·25% reprocessing losses. Physics parameters have been digital computed using the TDC (52) code, cylindrical geometry, inserting ANL 11 group cross-sections.

Assuming capital, operation and maintenance and insurance costs totalling 0·25 d/kWh at 0·7 load factor, it is seen that about 2½% burn-up is necessary to achieve reasonably economic power of about 0·45 d/kWh. Somewhat less would be acceptable if these high reprocessing and refabrication costs could be reduced.

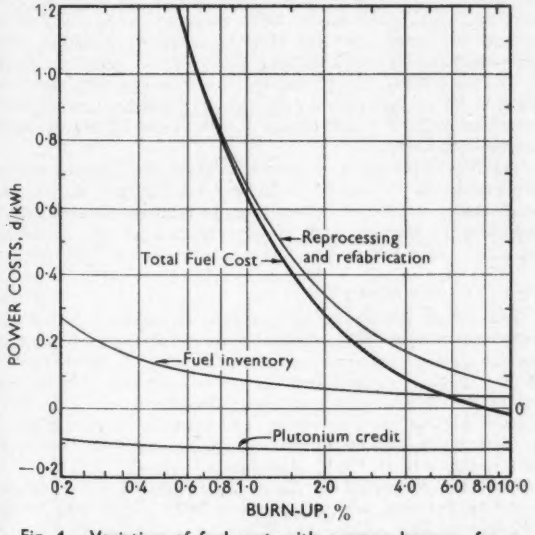


Fig. 1.—Variation of fuel cost with average burn-up for a 750 MW(th) fast reactor (see Table 4, column h).

Safety and Fuel Specification

The fast reactor then is justified economically, only if a fuel burn-up of about 2½% can be achieved at a power density of about 100 W/g, with a centre temperature of at least 650°C with metal fuel, and preferably above 1 200°C with UO_2-PuO_2. Compared with present thermal reactor practice, the fast reactor demands 10 times higher burn-up at 3 to 30 times higher specific power, and at higher centre temperature. Moreover, coolant channels tend to be smaller and a blocked coolant channel could lead to progressive fuel melting; this could lead to prompt criticality, possibly involving a nuclear explosion sufficient to damage the plant severely. Thus, the highest safety standards have to be maintained, despite the onerous irradiation conditions. On the credit side, the fast reactor is tolerant of the amount and type of fuel diluent or cladding material so far as poisoning is concerned. However, too generous fuel dilution is a potential safety hazard, as it can lead to increased fuel enrichment and to a larger potential excess reactivity on meltdown. It also makes for a smaller delayed neutron fraction, which complicates control and reduces safety factors, larger reactivity changes with burn-up, less U^{238} fission and reduced breeding.

Safety problems in the fast reactor arise from the large potential excess reactivity. If steel and sodium were removed from the core and the blanket moved inward leaving no core voidage, the number of critical masses so formed would be from 10 to 36, depending on the core design and the fuel chosen. This is an approximate indication of the theoretically worst case, and the number of critical masses thus calculated gives a relative assessment of the ultimate safety of the fuel. In a conceivable accident sequence, the number of critical masses is more likely to be between 1 and 2, but nevertheless the chance of prompt criticality must be treated with great seriousness. This is the distinguishing feature of the fast reactor; in a thermal reactor, gross concentration of fuel would lead to a reduction of reactivity. The size of a nuclear explosion following a given excess reactivity is aggravated by the short fast-neutron lifetime of 0·1 to 1 μs compared with 100 μs in a light water reactor and 1 ms in a graphite-moderated reactor; apart from this the short fast-neutron lifetime is unimportant and it does not affect the controllability.

The dependence of the delayed neutron fraction, β, on the type of fuel is indicated in Table 2 and the reactivity effects of different disturbances in Table 3. The net effective reactivity is the initial disturbance plus the feedback effects due to reactivity changes with temperature and temperature gradient, mostly due to thermal expansion effects. Typically the power coefficient of reactivity is 0·002% reduction per 1% increase in power at full power and flow, associated with a time delay of about one to two seconds. It is seen that the chance of an explosion is strongly influenced by β; it controls both the chance of an accident leading to fuel melting and the chance of molten fuel going critical. Fortunately the excess reactivity held up in a fast reactor core can be kept low at about 0·25% minimum, compared with 10 to 15% in a large gas-cooled reactor; this goes a long way to offset the danger of the large potential excess reactivity on fuel melting, by reducing the likelihood of this event.

From the previous discussion and Table 2, it would appear

* "The author was, until recently, Fast Reactor Technology Manager at the Dounreay Experimental Establishment and this article is the product of his experiences there, although the views expressed do not necessarily coincide with those of the U.K.AEA. Digital computing was performed by the computing service at Risley."

reasonable to conclude that the low delayed neutron fraction, β, of plutonium is a potential hazard and it is desirable to supplement it with U^{238} fast fission. In addition, the prompt negative power-coefficient of reactivity should be enhanced, if necessary by special flux or temperature sensitive devices in the core.

TABLE 1.—Capital costs of proposed 750 MW(th) reactor compared with Enrico Fermi

	Enrico Fermi	Proposed design	
	10^6	10^6	£10^6
Reactor building and shielding	2·3	2·3	0·58
Reactor	4·5	1·4	0·35
Fuel handling	3·4	1·4	0·35
Primary system	4·5	3·5	0·88
Secondary system	2·8	3·3	0·82
Instrumentation, control room	1·4	2·0	0·50
Auxiliary facilities	4·6	3·5	0·88
Total reactor cost	23·5	17·4	4·4
Supervision, design, conting., interest during construction	13·9	10·5	2·6
Power house (incl. turbo-gen.)	10·5	14·0	3·5
Supervision, design, conting., interest on above	3·5	4·6	1·2
Site and services	2·0	2·5	0·6
TOTAL	53·4	49·0	12·3
R & D, CEGB charges, etc.	5·4	5·0	1·2
Core and blanket fabrication costs	—	1·5	0·4
TOTAL CAPITAL COST	58·8	55·5	14.0
Cost/kW Installed	$392	$222	£56
In round terms	$400	$240	£60

COMPARISON OF FUELS

The main requirements then of a fast reactor fuel are:—

(1) It must be capable of reliable high burn-up at high specific power and high centre-temperature, with no fuel element failure at temporary overloads of 150% full-load.
(2) The proportion of fertile material needs to be high (enrichment low) for maximum fertile fissions, maximum breeding, minimum reactivity change with burn-up and maximum delayed neutron fraction.
(3) Good compatibility between fuel can and coolant, preferably to permit 500°C average coolant outlet temperature and safety at overloads of 150% full-load.
(4) The fuel should be cheap to process and fabricate.
(5) The amount of oxygen, carbon or similar moderating material should be as low as possible to prevent spectrum degradation.
(6) Fissile fuel should have no tendency to segregate out on fuel melting.

Table 4 summarizes reactor details using different fuels, to bring out the important factors quantitatively. Each reactor design is realistic both in likely burn-up and main design features. To simplify comparison, each reactor is of the same power density (25 kW/in³ fuel alloy) and the same reprocessing and refabrication cost is assumed (£2 700/l alloy). Physics parameters are digital computed using spherical geometry and ANL 11 group cross-sections. Burn-up is estimated from known data where this exists and by the method of Blake[3]; 2% diametric expansion of the fuel pin is assumed to be the maximum burn-up criterion.

(a) Uranium-10 w/o Mo_2

This fuel was chosen for the first Enrico Fermi charge and a similar 5 w/o fissium fuel in a stronger can is to be used in EBR II. The present author considers that under fast reactor conditions, making allowance for the higher thermal stresses and high safety factor necessary, a burn-up of no better than 0·3% can be hoped for. The EBR II approach of trying to achieve 1% burn-up, with rapid, low cost, on-site reprocessing and refabrication, appears the only road to success with this fuel.

TABLE 2.—Delayed neutron fraction values

Fuel	β, %	Complete reactor	β, %
Plutonium–239	0·21	100 MW(th), Pu metal fuel	0·37
Plutonium–240	0·26	750 MW(th), Pu metal fuel	0·39
Uranium–235	0·65	750 MW(th), U^{235} metal	0·72
Uranium–238	1·57	750 MW(th), Pu–s.s. cermet	0·28

The main reason for poor irradiation stability appears to be that the fuel relies on its own strength to resist swelling. This faces the difficulties that high strength and high temperature operation are incompatible; irradiation induced stresses and thermal stresses, due to high power-density, are additive, the fuel skin-stresses being particularly high; the fission-product gases collect in irregular-shaped cavities, so making the fuel a very inefficient pressure vessel; the large dimensional changes associated with phase-changes are superimposed and worsen irradiation stability; even with 10 w/o Mo added to uranium, its fuel creep-strength is still small in comparison with that of nickel-chromium and nickel-chromium-steel alloys.

(b, c) Stainless-steel Cermets

Cermets containing fuel in the form of oxide have been successful in enriched thermal reactors, and Enrico Fermi have selected it for their second charge (25 v/o UO_2) to effect a financial saving. The low thermal conductivity of this fuel encourages operation with a large temperature difference between fuel centre and surface to keep the pin-size moderate. In Table 4, a centre-fuel temperature of 650°C and an outlet temperature of 400°C is suggested. However, this promotes a high thermal stress of 55 klb/in², 0·17% strain, with fuel (b). A large safety factor of, perhaps, five appears necessary to allow for this, and also for possible inconsistent irradiation performance and poor behaviour on 150% overload, when a fission-gas release in excess of the 10% assumed is possible. The economics of this fuel appear to justify Enrico Fermi preferring it to 10 w/o Mo-U alloy, but the low breeding and disadvantageous safety features (low β, high enrichment, large number of critical masses) would appear to rule it out for plutonium as a long-term solution.

TABLE 3.—Reactivity changes associated with disturbances

Disturbance	Reactor type	Reactivity change
Loss of NaK	DFR	−5%
Sudden reduction of inlet temp. by 100° C	E.F.	+0·09%
1 in³ of fuel moved to reactor centre from edge of core	E.F.	0·006%
One control rod suddenly ejected (immediately following reloading)	E.F.	0·36%
Whole core and blanket moves inwards by 0·01 in at core edge	E.F.	0·1%
Sub-assembly added (61 per core) Central	EBR–II	2%
Sub-assembly added (61 per core) Peripheral	EBR–II	0·36%
1 pint of water added at reactor centre	DFR	1·1%
1 pint of Na removed from core	E.F.	−0·02%
1% gas entrainment	E.F.	−0·45%
1% power increase		−0.002%

The 50 v/o cermet is of interest because of its better physics properties, but the matrix is badly weakened and the thermal conductivity is lowered. This leads to reduced pin diameter, a reduction in burn-up and greater variation of the maximum burn-up. As a result, this fuel has little to commend it.

(d, e) UO_2 in a Strong Can

Two design approaches are possible: to accept a high centre temperature with 100% fission gas release, as in (d); or to restrict centre-temperature and hope for $< 10\%$ gas-release as in (e). Table 4 suggests that the second approach is better, but makes no allowance for increased fabrication costs due to the smaller pin diameter; moreover, the increased factor of safety in the burn-up estimate may not discount adequately the danger of relying on $< 10\%$ gas-release. Good fuel/can and can/coolant compatibility places little restriction on coolant outlet temperature, which is raised to 500°C, improving station efficiency to 36%.

A fear with ceramic fuels in strong cans is that a can defect

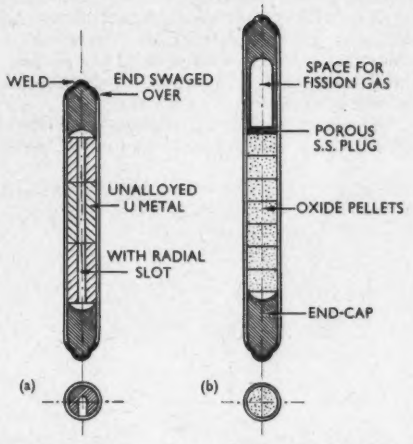


Fig. 2.—Fuel capsule used in irradiation experiments. The fuel diameter is 0·10 in and the cans are Nimonic 80 of 0·10 in wall thickness. Total length of fuel is approx. $\frac{3}{4}$ in.

could lead to ingress of coolant on shut-down, to be followed by vaporization on start-up and can bursting[6], as there is no hope of operating UO_2-PuO_2 at less than the sodium boiling point, 880°C. Reactors (d) and (e) respectively have 23 000 pins totalling 16 miles, and 32 000 totalling 41 miles; the chance of a defect is high, therefore. The exemplary standards of safety would appear to demand copper-bonded duplex-cans, or a low power density so that the occasional burst can will not lead to runaway fuel melting.

(f, g) UC in a Strong Can

Irradiation cracking will impair the high thermal conductivity of UC, but even assuming a 3 : 1 reduction, it is still high compared with UO_2. This is responsible for its improved performance in Table 4, but higher processing and fabrication costs have yet to be taken into consideration. The previous comments relating to (d) and (e) also apply, particularly the danger of can defects and reliance on low fission-gas release.

(h, k) Fuels Containing Void-space in a Strong Can

The author has previously elaborated his arguments[3], that, if fission gas accumulation is the chief cause of fuel swelling, as would appear from post-irradiation examinations reported[7,8], the burn-up could be extended considerably by providing room for the gas within the can, which should be as strong as possible. This theory has been tested in the Dounreay Materials Testing Reactor under representative fast reactor conditions[4]. Three fuel-pins of unalloyed-uranium in Nimonic 80 cans (at 550 W/g, 700°C centre temperature, 500°C surface) were taken to 2·7% burn-up; two at similar ratings were taken to 4·7% burn-up, and two of UO_2 in Nimonic 80 cans (1 600°C centre, 460° surface, 560 W/g) to 5·3% burn-up (see Fig. 2). In each case the void-space provided was about 30% of the can internal volume at room temperature. All specimens survived with negligible swelling or can distortion (Figs. 3-5); some were also tested at high temperature afterwards (the UO_2 specimens to 900°C) with and without simulated can ruptures, and survived. In the light of theory and these tests, the burn-up figures of Table 4 should represent what is reasonably achievable. Not only is burn-up high, but so is the permissible centre temperature; also effects of thermal stresses and phase changes in the fuel are unimportant. Moreover, the U^{238} content is high, bringing with it the many benefits in regard to safety previously mentioned.

Despite the higher coolant outlet temperature of the ceramic fuels (i, j) and a net efficiency of 36% compared with 31%, the total power costs between cases h, i, j are negligibly different, especially as the higher carbide fabrication costs have not been included, nor the fact that, with so large a pin, the can thermal stresses are high.

TABLE 4

Relative merits of different fuels in a fast power reactor (Standard case: 750 MW(th), 80% load factor, Pu ^{239} fuel, standard 50 cm blanket (1) and 25 cm thick Fe reflector, £2 700/litre fuel cycle cost)

	Strong fuel; weak can			Weak fuel; strong can				Voided fuel; strong can			
	a	b	c	d	e	f	g	h	i	j	k
Fuel composition, v/o; Pu+U^{238}	81·2	14·5	24·2	48·5	48·5	63·7	63·7	72·5	38·3	50·3	71·2
O or C	—	14·0	23·3	46·5	46·5	31·3	31·3	—	36·7	24·7	8·8
Diluent	16·8	66·5	47·5	—	—	—	—	—	—	—	—
Void	2·0	5·0	5·0	5·0	5·0	5·0	5·0	27·5	25·0	25·0	20·0
Cladding	Zr	s.s.	s.s.	Nimonic 80				Nb/Nim	Nimonic 80		
Fuel dia., in	0·191	0·174	0·161	0·182	0·151	0·238	0·160	0·194	0·182	0·238	0·194
Thickness/dia.	0·05	0·05	0·05	0·1	0·1	0·1	0·1	0·1	0·1	0·1	0·1
Recommended max. temp, °C; fuel centre	600	650	650	2 100	1 200	1 800	1 000	700	2 100	1 800	700
fuel surface	483	459	443	587	564	640	571	547	587	640	547
coolant	392	392	392	492	492	492	492	442	492	492	442
Estim. max. burn-up, % fuel	0·6	34	7	5·4	18	5·9	18	14	27	22	25
Safe max. burn-up, % fuel	0·3	7	1	2	4·5	2	4·5	6	11	9	9
Safe max. burn-up, MWd/l (alloy)	43	180	42	173	386	226	508	772	750	800	1 140
F.P. gas release, %	100	10	10	100	10	100	10	100	100	100	10+18
Core specific power, W/g (fuel)	99	553	332	165	165	126	126	111	210	159	113
Days in reactor	35	149	35	141	318	185	417	630	613	660	934
Coolant inlet temp, °C	200	200	200	300	300	300	300	250	300	300	250
Temp. outlet, °C	400	400	400	500	500	500	500	450	500	500	450
Net station efficiency, %	27	27	27	36	36	36	36	31	36	36	31
Breeding ratio	1·70	·116	1·25	1·42	1·42	1·52	1·52	1·65	1·39	1·50	1·65
U^{238} fast fission	0·233	0·122	0·144	0·184	0·182	0·206	0·202	0·222	0·175	0·195	0·22
Delayed neutron fr., %	0·38	0·28	0·29	0·33	0·33	0·35	0·35	0·37	0·32	0·34	0·37
No. of critical masses	10	36	24	13	16	9	12	13	20	17	13
Plutonium prod., g/MWd	0·694	0·204	0·300	0·476	0·463	0·576	0·558	0·660	0·443	0·538	0·660
Reprocessing losses, g/MWd	0·110	0·043	0·160	0·045	0·035	0·041	0·034	0·032	0·031	0·031	0·030
Net Pu prod. kg/a 219(m−1)	128	35	31	94	95	117	115	137	90	111	138
Net Pu prod., % /a 100μ/M_1	3·6	4·4	1·3	9·6	12·5	13·8	16·3	16·3	11·9	14·5	17·0
Reactor capital+Oper. and Main, d/kWh	0·31	0·313	0·31	0·268	0·268	0·268	0·268	0·290	0·268	0·268	0·290
Fuel capital costs, d/kWh	0·20	0·045	0·13	0·041	0·032	0·036	0·029	0·041	0·032	0·032	0·039
Reproc. and refab., d/kWh	2·32	0·555	2·38	0·433	0·195	0·332	0·148	0·113	0·100	0·094	0·076
Pu production, d/kWh	0·13	0·036	0·03	0·072	0·089	0·089	0·088	0·121	0·069	0·085	0·122
Total power cost, d/kWh	2·7	0·88	2·8	0·67	0·42	0·55	0·36	0·32	0·33	0·31	0·28

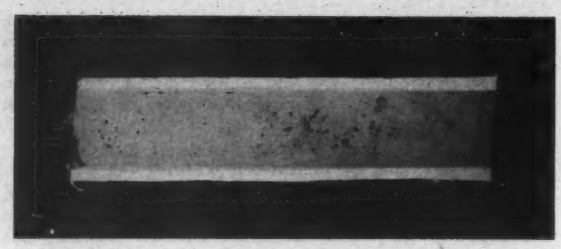


Fig. 3—Micrograph section of a fuel pin taken to 2·7% burn-up. Irradiation 550 W/g, 700°C centre, 500°C surface. Longitudinal section ground to 0·025 in from centre.

The only fuel significantly better than unalloyed uranium is the 18 v/o UO_2 or UC in a voided matrix of uranium, but this relies on the unproven assumption that $< 10\%$ fission-gas release in the ceramic is assured; furthermore, it does not take account of higher fabrication charges, nor the safety problem associated with the possibility of plutonium segregation on accidental melt-down. However, method (k) does leave the plutonium in the relatively inert oxide or carbide phase, which permits Nimonic canning material to be retained.

FUEL CHOICE

It would appear most reasonable, therefore, at this stage to concentrate attention on the simplest of the fuels: U-Pu alloy in a strong can. The canning material could be Nimonic if, at the appropriate enrichment, the fuel/can compatibility is acceptable. If it is not, a high creep strength niobium-alloy can[9] could be used, or a duplex-can of niobium-alloy inner and Nimonic outer, or U^{238} inner and Nimonic outer. Improvements attainable with PuO_2/UO_2-U cermet may provide a possible future bonus; so can raising the coolant outlet to 500°C. Metallic fuel is the best known of all fast reactor fuels and is used in ERB I and II, Enrico Fermi and the Dounreay fast reactor.

In all these arrangements a doubt persists of the fuel pin behaviour with a defective can. DMTR tests[4] have helped to restore confidence that a can defect will not result in fuel extruding and blocking a coolant passage. Tests have been made with irradiated pins in a furnace, with a hole drilled through the can wall. There was little tendency for the fuel to extrude. However, a can defect could still permit ingress of coolant on shut-down, and subsequent fuel expansion may seal it in; this would remove some expansion space and reduce maximum burn-up, making can-swelling likely. To prevent this leading to trouble, the simplest approach would be to restrict fuel-rating and employ well-interconnected coolant passages, so that a blocked coolant passage would not initiate runaway fuel meltdown if the economic penalty of this approach proves acceptable.

TABLE 5.—Likely power cost breakdown in the power fast reactor

Load factor	0·7	0·8
Capital charge, d/kWh	0·194	0·170
Operation, maintenance and insur.	0·060	0·060
Fuel charge	0·062	0·060
Total electricity cost, d/kWh	0·32	0·29

The introduction of void within the fuel opens up new possibilities. In the DMTR tests, for example, the void was mainly introduced as a radial slot to prevent thermal stressing of the can by the fuel, and to amplify diametric expansion of the fuel. Provided the mating surfaces are clean and of high surface finish (<30 μin), an intermediate bond between fuel and can may be avoided, which is a great simplification. It also solves the problem of achieving a reliable thermal bond, as sodium-filling cannot yet be regarded as successful.

The amount of void may be varied within the core, depending on the local temperature and rate of burn-up. More important, the method can equally be applied to the blanket, for which a dense, high burn-up fuel is also necessary. In an inner-blanket sub-assembly, the power-density can vary by as much as 10^3: 1. Thus, the variation of voidage from a high level (say 20%) near the centre-plane, to a few per cent at the blanket outer surface is very beneficial in equalizing burn-up and improving the blanket's effectiveness. The axial blanket power density also varies similarly, and a small diameter rod of high voidage is required near the core, and a larger solid rod farther away.

Successful fast reactor operation would appear possible, therefore, with a simple fuel pin arrangement of unalloyed U-Pu metal, containing 25-30% of the internal can volume as void-space, in the form of a radial slot. This can be spiral or axial,

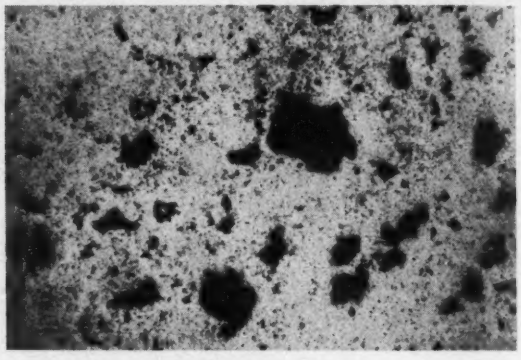


Fig. 4.—Pin irradiated as for Fig. 3. Area of macro porosity in section ground to centre.

with the slot position staggered from fuel pellet to pellet. The voidage should be less at the inlet and greater at the outlet end of the fuel-pin. A can of wall thickness of about 10% of the can diameter is desirable.

Retention of fission-gas within the can is the more straightforward engineering solution at this stage, although venting fission-gas to the coolant shows possibilities, even with metallic fuel, provided the voidage fraction is large enough. It would have advantages of very high burn-up capability, low can pressure and would be more accommodating of can defects.

Reactor Description

A section of a 750 MW(th), 250 MW(e), power fast reactor, as proposed by the author, is shown in Fig. 6. It attempts to achieve a higher standard of safety than existing fast reactors, by the following features:—

(1) The specific power is 50 to 70 W/g, compared with 150 W/g with EBR II and Enrico Fermi. This permits more rugged pins to be employed of 0·25 to 0·28 in o.d. Together with fuel pins on a triangular lattice with well-interconnected coolant channels, it ensures that a complete coolant-channel blockage will not produce runaway fuel melting, even at 150% full power. The sodium velocity is kept a little low at 25 ft/s, which helps in this respect, and also ensures adequate natural convection cooling on shut-down. At the same time this is not far away from the economic optimum velocity of the coolant.

(2) Excess reactivity in controls is limited to less than \$0·5 at full power and \$0·8 at shut-down. Control rod reactivity is

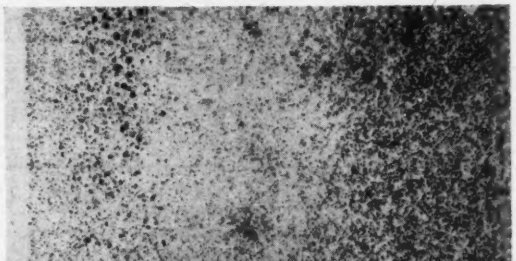


Fig. 5.—Pin irradiated as for Fig. 3. Variation from can wall to centre.

monitored at all times. Limitations on rate of change of reactivity on the other hand are less severe than current practice.

(3) The core is restrained at the mid-plane with an inward pressure from the inlet coolant and by special pressure-applying sub-assemblies within the blanket. Flux-flattening is not employed, except axially by fuel voidage variation and here more to flatten maximum burn-up capability. Coolant gagging is introduced in the core, sufficient to reduce sub-assembly outlet temperature variations by half the amount of the ungagged case.

(4) Eight core pumps and eight blanket pumps in parallel are used, associated with three to four intermediate heat exchangers. Electromagnetic pumps are used in preference to mechanical, as their lower capital cost, smaller size, greater reliability and freedom from maintenance, absence of mechanical drives and freedom in location, more than offset their lower efficiency. Moreover, many pumps in parallel are practical, which is highly desirable to guard against loss of flow and to enable low-flow warnings to be made more reliable. These are the same safety reasons for their selection in DFR, but in the present design, by arranging pumps in parallel and using a few shell-and-tube primary heat exchangers, the complication and expense of the DFR arrangement are avoided.

(5) All sodium circuits are trebly contained with inert gas in both interspaces. The only sodium/cover-gas interfaces are in the expansion tanks, and these have stainless-steel floats, which virtually eliminate the interface area.

(6) A simple core-dispersion device is incorporated, to lift sub-assemblies from the core to thimbles outside the reflector. The core is dispersed whenever treble containment is violated, such as to remove a pump, or at times of potential danger which could lead to fuel melting.

(7) The containment cylinder is kept compact (60 ft dia. × 120 ft), and below ground with missile protection, on the basis that the world is best protected from the fast reactor by protecting it from the world.

(8) An 8-in stainless steel reflector surrounds the blanket, which keeps the blanket thickness down to 14 in or three rows of sub-assemblies. A complete row is unloaded at a time, and outer rows moved inward, to preserve the power pattern and to avoid the need for orifice control of individual sub-assemblies during operation. Blanket elements are taken to 1 to 2% burn-up at the centre plane.

(9) An EBR II tank-type construction is employed, with a simpler shielding arrangement, whose weight is kept off the primary tank.

(10) There is extensive use of thermal insulation within the primary tank, with reactor inlet pipes insulated externally and the reactor vessel and outlet pipes internally. Thus the tanks are maintained isothermal and the reactor can sustain full scrams from power. This eliminates the need for power set-backs. Set-backs or slow scrams can make an accident worse, as the need to overcome the set-back to initiate the full scram conflicts with the requirement for a simple, unambiguous safety system.

(11) A built-in reloading-cave with windows is incorporated to simplify and speed up reloading, and to preserve treble containment; also to give better blast-protection.

Despite the additional safety features, the layout attained is much simplified over existing fast reactors, which should be reflected in substantially lower capital cost. Despite its large output of 750 MW(th), its containment building is smaller than any existing fast reactor and is only 30% of the volume of the 60 MW(th) Dounreay fast reactor.

CONCLUSIONS

The case is made here that the fast reactor can be built for a capital cost as low as £60/kW installed, with standards of safety significantly better than any existing fast reactor and adequate for a large-scale power fast reactor programme. Tests with unalloyed uranium to nearly 5% burn-up under conditions appreciably worse than in a power fast reactor suggest that 6% average burn-up can be achieved with uranium/plutonium metal fuel at a power density of about 50 to 100 W/g, with a maximum centre temperature of 700°C and with a coolant average outlet temperature of 450°C. These conditions of coolant outlet temperature and power density are conservative with respect to current fast reactor practice, but the centre temperature and the burn-up are very much higher.

Assuming £60/kW net, and 5·5% interest charges and 20 years reactor life, giving 8·37% capital charges, the electric power costs of the fast reactor build up as in Table 5, where it can be seen that at 0·7 load factor, power would be produced at 0·32d/kWh.

To achieve this does not require a significant extrapolation of present-day knowledge and furthermore it does not represent the ultimate capability of the fast reactor. Capital costs could fall to £50/kW in large plant; probably a fivefold increase in burn-up is practicable based on venting fission gas; and increases can be expected in outlet temperature and steam plant efficiency. On the other hand, the estimated power cost is well below that of any other nuclear or conventional plant, and gives a large contingency in case of error or unforeseen difficulty. Thus, £70/kW and 3% average burn-up could still be competitive in the United Kingdom.

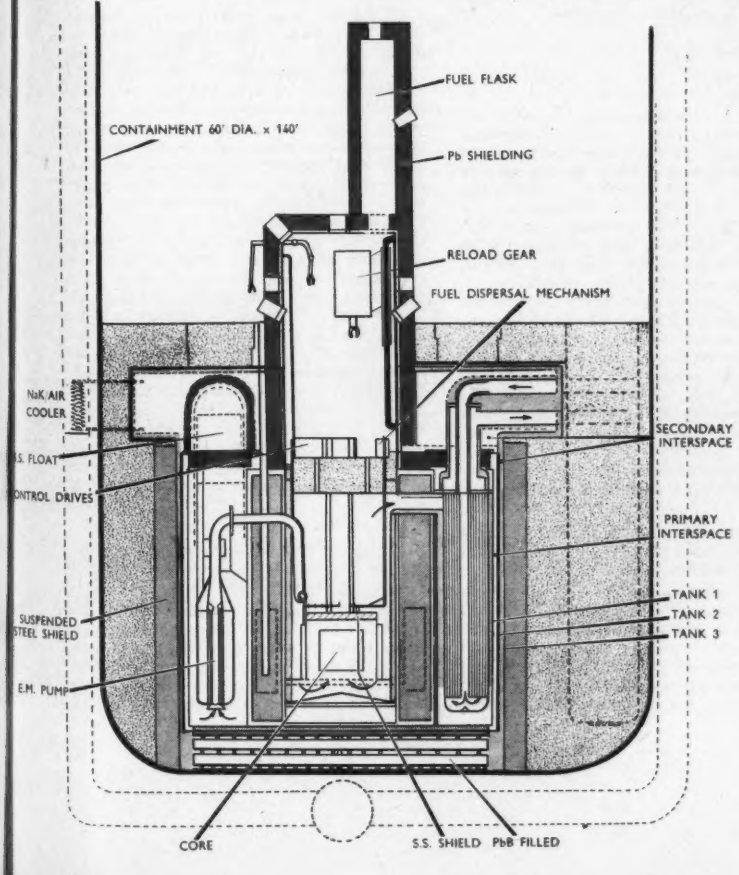


Fig. 6.—Vertical section through the containment building of the proposed 750 MW(th) power fast reactor.

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TRANSLATIONS

D'Hinkley à Sizewell (364)

La centrale nucléaire d'Hinkley Point est al première des centrales nucléaires du Central Electricity Generating Board dont la puissance atteint 500 MW. Elle est construite par le consortium English Electric, Babcock and Wilcox, Taylor Woodrow and Atomic Power Company et devrait etre mise en service d'après les dernières estimations en automme 1963. La principale cause du retard dans la construction est due aux changements apportés au coeur à la suite des progrès réalises récemment dans la connaissance du graphite ainsi que diverses difficultés inattendues recontrées lors de la construction. Il est tenu largement compte de tous ces éléments nouveaux lors de la préparation du projet de la centrale de Sizewell dont la construction doit également être effectuée par le consortium. Le montant des investissements par kW installé devrait être beaucoup plus bas pour cette dernière centrale que pour celle d'Hinkley Point. Les principales raisons de cette différence sont discutées ici.

Le coût du combustible pour les réacteurs rapides (369)

Des investissements et un coût du combustible peu élevés devraient permettre au réacteur rapide de produire de l'énergie électrique à meilleur compte que n'importe quel réacteur thermique à condition que des taux d'utilisation du combustible beaucoup plus élevés que les taux actuels puissent être obtenus. Suite à une étude comparative de divers combustibles on arrive à la conclusion que l'alliage uranium-plutonium métalique ou à base d'oxyde comportant un espace vide dans une gaine résistante offre les meilleures perspectives en vue de l'obtention de taux de combustion élevés dans de bonnes conditions de sécurité. Des essais obéissant au principe précédent ont été effectués par l'auteur. Un taux de combustion de l'ordre de 5% a pu être obtenu sans qu'il en résulte un fluage appréctable de la gaine. Avec un tel combustible, une quote-part de 2 mil/kWh semble raisonnable pour le coût du combustible.

Enrico Fermi (380)

Le réacteur rapide Enrico Fermi sera le premier que aura été construit principalement pour la production d'énergie plutôt que dans un but expérimental. L'expérience en matière d'exploitation acquise avec cette centrale contribuera beaucoup à éliminer certaines incertitudes se rapportant à l'exploitation des réacteurs rapides. Construit pour un groupe de compagnies américaines du secteur publique, le réacteur chargé initialement d'un alliage d'uranium et de molybdène est réfrigéré au sodium à l'aide de trois circuits. La chaleur récupérée dans ceux-cis est transmise à un circuit secondaire, au sodium egalement, dans un échangeur de chaleur tubulaire. Elle passe finalement au circuit tertiaire dont la vapeur alimente les turbines. Une description complète des installations est donnée. Elle comporte des détails relatifs à la construction du coeur, ainsi qu'à la fabrication des éléments de combustible, du dispositif de manutention de ces éléments, des circuits primaire et secondaire, des systèmes de contrôle et de sécurité.

Amenagement du coeur du réacteur rapide de Dounreay sans perturbation de l'exploitation (386)

Pour pouvoir permettre le changement facile du coeur du réacteur de Dounreay, il a été nécessaire de mettre au point un appareillage capable de fonctionner dans le sodium liquide.

Von Hinkley zu Sizewell (364)

Das Atom-Kraftwerk Hinkley Point ist das Erste von den 500 MW-Kraftwerken des Central Electricity Generating Board. Es wird von der English Electric, Babcock & Wilcox, Taylor-Woodrow Atomic Power Company gebaut, und es ist jetzt geplant, dass es im Herbst 1963 in Betrieb genommen wird. Die Hauptursache der Verzögerung in der Vollendung des Baues sind die Aenderungen gewesen, die mit dem Kern vorgenommen wurden, die sich weiter aus den neuen Daten für Graphit ergaben, es sind aber auch einige der bei der Konstruktion auftretenden Probleme aufgeführt worden als Hintergrund für eine Zusammenfassung der Haupt-Charakteristiken der Konstruktion von Sizewell, für die die Company ebenfalls verantwortlich ist. Die Kapitalkosten per Kilowatt für dieses Werk liegen erheblich unter denen von Hinkley, und die hauptsächlichen Gründe dafür werden diskutiert.

Die Brennstoff-Wirtschaft bei den schnellen Reaktoren (369)

Niedrige Brennstoff-und Kapitalkosten sollten den schnellen Reaktor dazu geeignet machen, elektrische Energie billiger zu erzeugen als jeder thermische Reaktor under der Veraussetzung, dass ein vielmals grösserer Abbrand erzielt werden kann als die derzeitige thermische Praxis zulässt. Aus einer Uebersicht über die verschiedenen Brennstoff wird der Schluss gezogen, dass Uran-Plutonium als Metall oder Oxyd mit vorgesehenem leeren Raum in einem starken Mantel für das Element die besten Aussichten für hohen Abbrand bei voller Sicherheit bietet. Das Prinzip ist von dem Autor in Versuchen geprüft worden, wobei ein Abbrand bis rd. 5% erzielt wurde, ohne dass der Mantel messbare Erweiterungen zeigte. Mit einem derartigen Brennstoff scheinen die Kosten einer Brennstoff-Periode 2 mil/KWh durthaus annehmbar.

Enrico Fermi (380)

Enrico Fermi wird der erste schnelle Reaktor sein, der in der Hauptsache für die Produktion von Energie und nicht nur für private Versuche gebaut wird. Betriebserfahrungen, die man in dem Werk sammeln wird, werden in hohem Masse dazu helfen Ungewissheiten zu beseitigen, die sich aus dem Betrieb eines schellen Reaktors ergeben. Der Reaktor, der für eine Gruppe von amerikanischen Gesellschaften für öffentliche Betriebe gebaut wird, wird mit Natrium in drei Kreisläufen gekhilt; die Wärme, die von diesen abgeführt wird, geht durch einen sekundären Natrium-Kreislauf mit Röhrenkessel-Wärmeaustauschern und von diesem sekundären Umlauf in den Dampf-Umlauf, von dem die Turbinen gespeist werden. Es wird eine umfassende Beschreibung der technischen Seite des Werkes gegeben, die Einzelheiten der Konstruktion des Kernes enthält, der Brennstoff-Elemente, der Handhabung des Brennstoffs, des primären Umlauf-Systems und des sekundären Umlauf-Systems und der Kontroll-und Sicherheits-Einrichtungen.

Das Auswechseln der Kernrohre im Schnellen Reaktor von Dounreay unter den Bedingungen der Aufrechterhaltung des Betriebes (386)

Um es möglich zu machen den Atomkern in Dounreay ohne grosse Schwierigkeiten auszuwechseln, ist es nötig gewesen, eine Apparatur zu konstruieren, die in flüssigem Natrium arbeiten kann, und ein System von Schleusen zu entwickeln, die es gestatten Teile des Kerns herauszuziehen und dafür neue Teile wieder einzusetzen.

Desde Hinkley a Sizewell (364)

La central de energía nuclear de Hinkley Point es la primera de las centrales de 500 MW de la Junta Central Generadora de Electricidad. La están construyendo la English Electric, Babcox & Wilcox, la Taylor-Woodrow Atomic Power Company y se estima ahora que entrará a prestar servicio en el otoño de 1963. La causa principal del retraso en terminarla ha sido modificaciones del núcleo resultantes de nuevos datos sobre el grafito, pero algunos de los problemas de construcción se hallan itemizados como fondo al resumen de las caracteristicas principales de la construcción de Sizewell, también de contrucción de la compañía. La inversión de capital por kilovatio para esta central is muy inferior a la de Hinkley y se discuten los motivos de esto.

Problemas economicos del combustible en reactores rapidos (369)

Los gastos reducidos de combustible y de capital debieran permitir la producción de energía eléctrica con reactores rápidos mucho más barata que con cualquier otro tipo de reactor termal siempre y cuando se pueda lograr un quemado muchas veces mayor del que se logra en la práctica termal corriente. Como resultado del estudio de varios combustibles se ha llegado a la conclución de que el metal o el óxido de uranio-plutonio conteniando un espacio vacio en un recipiente rebusto ofrece las mejores probabilidades de un alto régimen de quemado con seguridad. El autor ha llevado a cabo pruebas de este principio y ha logrado aproximadamente el 5% de quemado con hinchazón aparente del recipiente. Con un combustible semejante parece razonable un costo de ciclo de combustible de 2 mil/kWh.

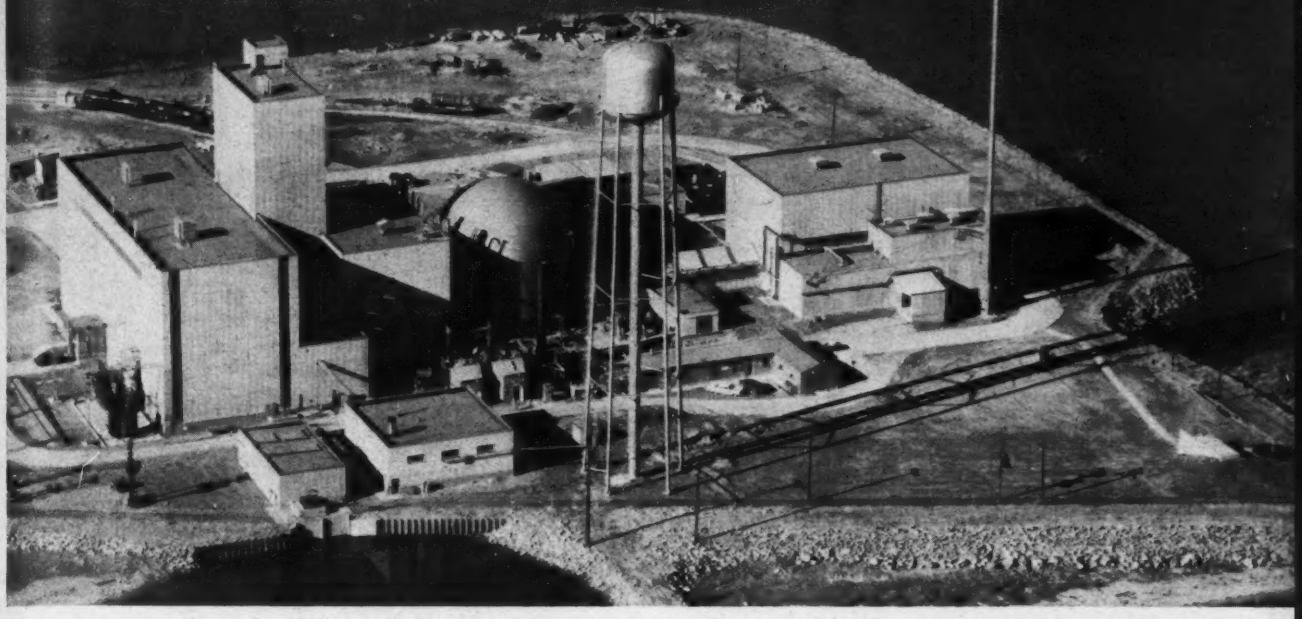
Enrico Fermi (380)

El Enrico Fermi será el primer reactor rápido construido especialmente para la producción de energía más bien que para experimentos particulares. La experiencia lograda en el servicio práctico con esta instalación contribuirá mucho a aclarar ciertas dudas relacionadas con el uso práctico de reactores rápidos. Construido para un Grupo de Compañías americanas de servicios públicos, el reactor, alimentado inicialmente con un combustible de uranio molibdeno, lleva enfriamiento por sodio en tres lazos, intercambiándose el calor de estos a un circuito secundario de sodio en intercambiadores de calor de casco y de tubo y desde este circuito secundario a un circuito de vapor que alimenta las turbinas. Se da una descripción bastante amplia de la mecánica de la instalación incluyendo detalles de la construcción del núcleo, elementos combustibles, manipuleo de combustible, sistema primario de circulación, sistema secundario de circulación y los mandos y las caracteristicas de seguridad. Este artículo va acompañado de un plano en perspectiva de la instalación en el cual se han encuadrado los parámetros principales de la instalación.

Para cambiar el grupo de tubos del nucleo de un reactor rapido Dounreay bajo condiciones de mantencion en servicio (386)

Para poder cambiar el núcleo Dounreay con la mayor facilidad ha sido necesario crear equipo capaz de trabajar en sodio liquido y un sistema de sifones que permita retirar secciones del nucleo e introducir otras nuevas. Se describe detalladamente la concepción de este equipo para Dounreay.

ENRICO FERMI



THE Enrico Fermi fast reactor plant at Monroe, Michigan, has attracted considerable attention over the past five years, for three reasons. First, as a fast breeder reactor, it is of outstanding technical interest; second, the scheme is backed by private capital, with a very large number of organizations participating; third, it has been, for almost the whole of the construction period, the centre of a nation-wide legal and political battle which, at one time, threatened to extinguish it altogether.

It is with the technical considerations that we are primarily concerned, but it will be as well to record briefly the responsibilities of the various participants in the scheme. The reactor plant was designed by Atomic Power Development Associates, Inc., a non-profit organization of 42 member companies, listed elsewhere. The nuclear portion of the plant is owned (and will be operated) by the Power Reactor Development Company, a non-profit organization, supported by 25 member companies which are also listed elsewhere, while the owner-operators of the steam side of the plant are the Detroit Edison Company. Architect-engineers for the nuclear portion of the plant are Commonwealth Associates, Inc., and construction engineers to both Detroit Edison and PRDC are United Engineers and Constructors, Inc.

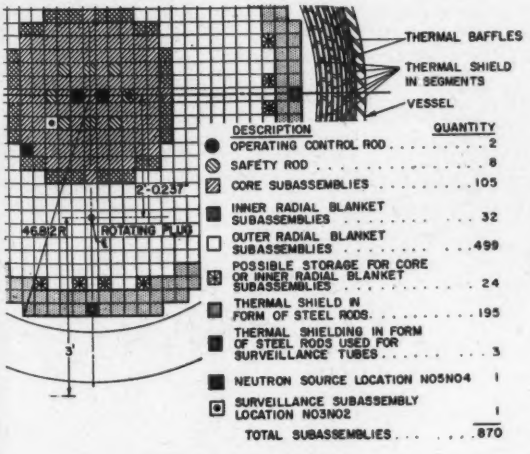

THERMAL BAFFLES
THERMAL SHIELD IN SEGMENTS
VESSEL
2'-0 23/32"
46 1/2" R
ROTATING PLUG
3'
DESCRIPTION
QUANTITY
OPERATING CONTROL ROD 2
SAFETY ROD 8
CORE SUBASSEMBLIES 105
INNER RADIAL BLANKET SUBASSEMBLIES 32
OUTER RADIAL BLANKET SUBASSEMBLIES 499
POSSIBLE STORAGE FOR CORE OR INNER RADIAL BLANKET SUBASSEMBLIES 24
THERMAL SHIELD IN FORM OF STEEL RODS 195
THERMAL SHIELDING IN FORM OF STEEL RODS USED FOR SURVEILLANCE TUBES 3
NEUTRON SOURCE LOCATION NO5NO4 1
SURVEILLANCE SUBASSEMBLY LOCATION NO3NO2 1
TOTAL SUBASSEMBLIES 870

Fig. 1.—Part section of core.

The plant is situated at Lagoona Beach, in Monroe County, Michigan, on the shores of Lake Erie, about 30 miles S.E. of Detroit. The site consists of about 915 acres, owned by Detroit Edison, the actual plant area being some 15 acres, of which 8 acres, as the reactor site, is leased to PRDC. The plant will have an initial electrical output of 60·9 MW net, although a turbine of 150 MW capacity is installed to take care of future increases in output. A brief description of the scheme, based on the information then available, was published in *Nuclear Engineering* for March, 1957, pp. 112-114.

The general arrangement of the plant can be seen from the pull-out drawing inserted overleaf, which shows a cut-away view looking from the west. The reactor enclosure containing the reactor vessel, intermediate heat exchangers and primary coolant pumps is approximately central. To the right are the steam generators, turbine house and control centre; to the left are the fuel and repair buildings and sodium service and gas treatment plants. A water tower and water treatment plant are located on the lake side of the reactor.

REACTOR

The reactor is a fast breeder type, with a thermal output from the first core of 200 MW, cooled by sodium with three coolant loops. The general construction is illustrated in the inset on the pull-out drawing, from which it can be seen that the reactor vessel is divided into two portions, the lower portion containing the core itself, and the upper portion, larger in diameter and offset from the lower portion, containing the hold-down mechanism and the offset fuel handling mechanism, both mounted in a rotating shield plug which provides closure for the vessel. Attached to one side of the upper vessel, and communicating with it, is the transfer rotor, for temporary storage of irradiated fuel elements; all these are again enclosed in a larger container, the primary shield tank.

Core

As a breeder, the Enrico Fermi core is divided into fissile and fertile or blanket regions. The elements in the central section are divided into three vertical zones so as to give axial blanket regions above and below the central fissile section. A cross-section of the core is shown in Fig. 1. The elements are

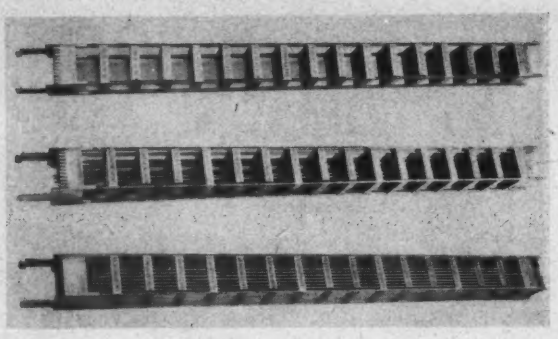


Fig. 2.—(Above) Fuel section of element.

Fig. 3.—(Right) Arrangement of fuel and upper and lower blanket sections of central elements.

arranged in a square lattice of 2·693 in pitch and space is provided for 871 fuel and blanket elements, control rods and thermal shielding bars.

Fuel Elements

The composite elements for the central zone are made up of three sections. The fuel section consists of 140 pins of uranium 10 w/o molybdenum alloy, the uranium being enriched to 25·6%, clad with zirconium. The original design envisaged 144 pins, with wire separators, but this was abandoned in favour of an "egg-crate" construction in which the pins are located by a series of stainless steel grids at 1⅞ in centres which are tied together at the corners with stainless steel pins replacing the four corner fuel pins. Each element has 10 grids which grip the pins by means of impressed dimples; seven grids merely guide the pins and restrict movement. The integrity of this design has been established by temperature cycle tests in flowing sodium; further tests using lead pins have established that the hydraulic forces would tend to straighten any bowed pins.

The upper and lower blanket sections are composed of 16 rods of uranium 3 w/o molybdenum alloy, the uranium being depleted to a total concentration of 0·35% U^{235}; these rods are also zirconium-clad.

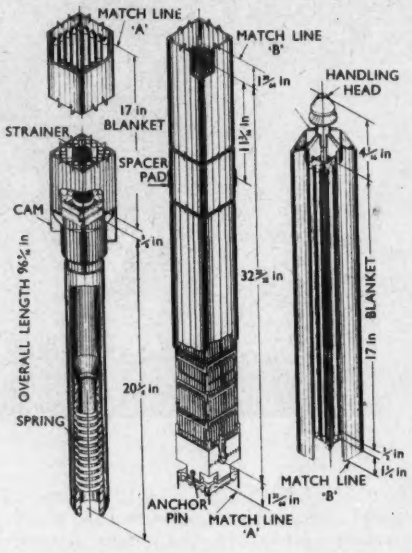
Radial blanket elements have 25 such rods running the full length of the element, and are contained in a stainless steel wrapper of approximately square section, with slightly radiused corners.

The inner blanket units (i.e., those immediately surrounding the core units) differ slightly from the remainder. As will later be shown, there are two distinct coolant flow rates, the velocity through the active portion being high enough to require positive hold-down of the elements against the flow. The inner row of blanket units are included in this system (so that they could be exchanged for active elements if required) and have, therefore, the same type of bottom fittings as the active elements, with springs to accommodate thermal expansion. The remainder of the blanket elements, operating with a much lower coolant velocity, are held down by gravity only, and, lacking top restraint, do not require expansion springs.

Fig. 5.—(Left) Looking down into the reactor vessel during erection.

Reactor Vessel

The lower reactor vessel, 9·5 ft diameter, provides accommodation for the core and blanket. About 4 ft from the domed bottom there is an annular box structure, serving to divide the coolant into two zones, a high-pressure central zone for the main coolant flow, and a low-pressure zone within the annulus itself. Above this is situated the core support proper, which consists of two 2 in plates spaced 14 in apart by welded ribs, and drilled to receive the lower end fittings of the fuel and blanket elements. The central (active core) plates are removable.



Below the level of the support structure, with their centre lines 2 ft 8 in from the domed bottom, are three 14 in nozzles

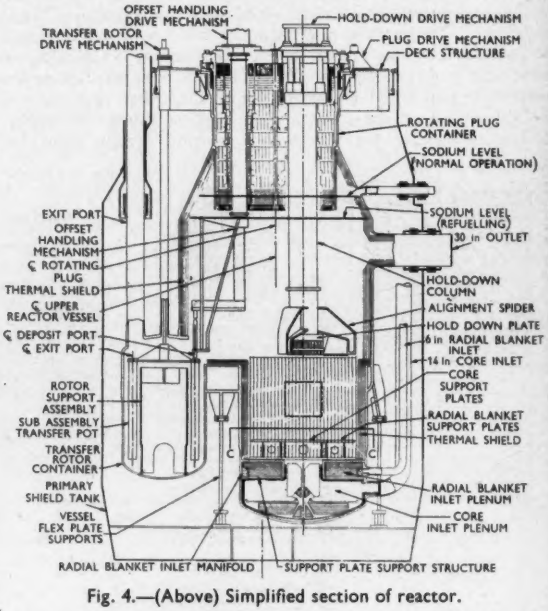
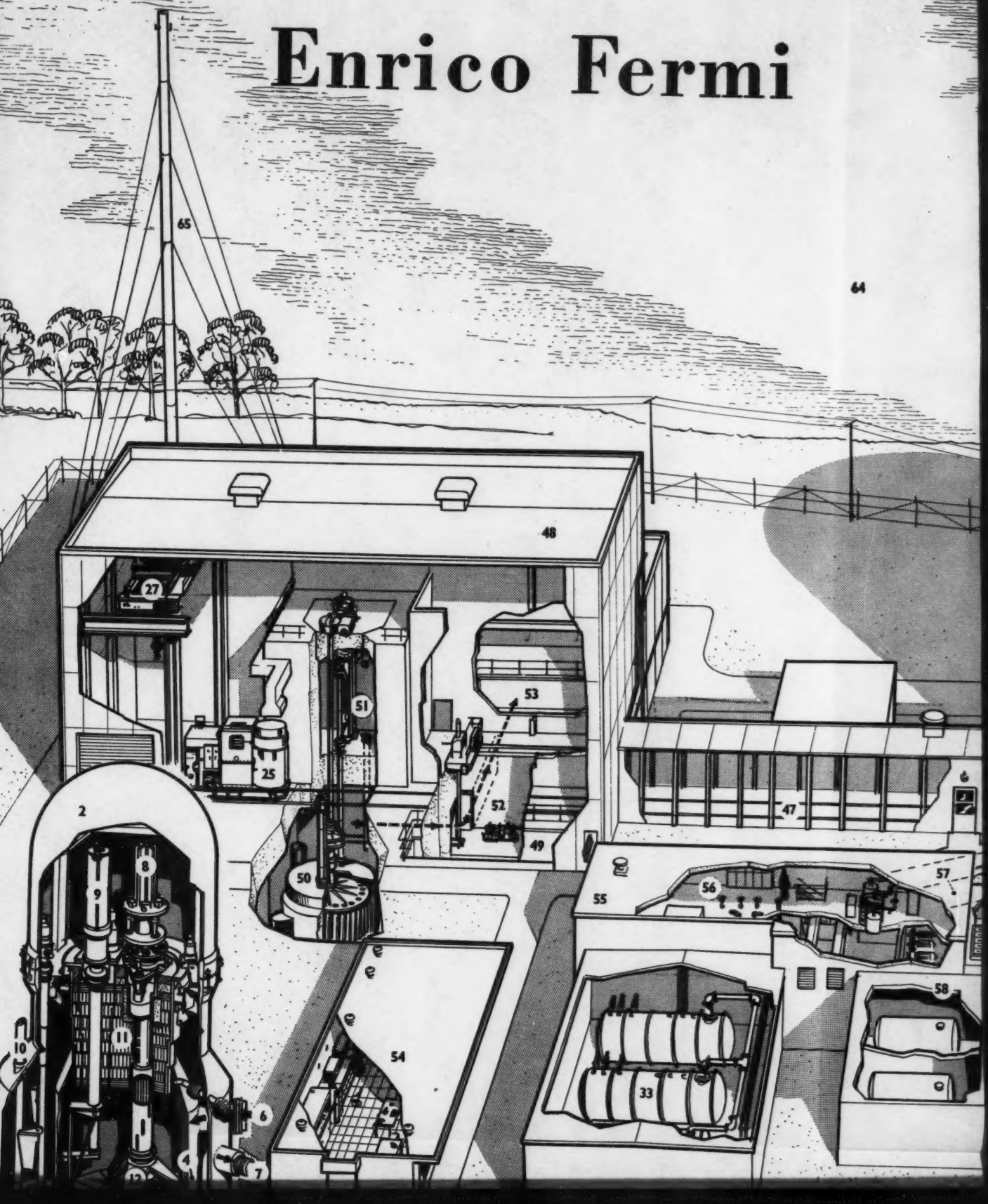
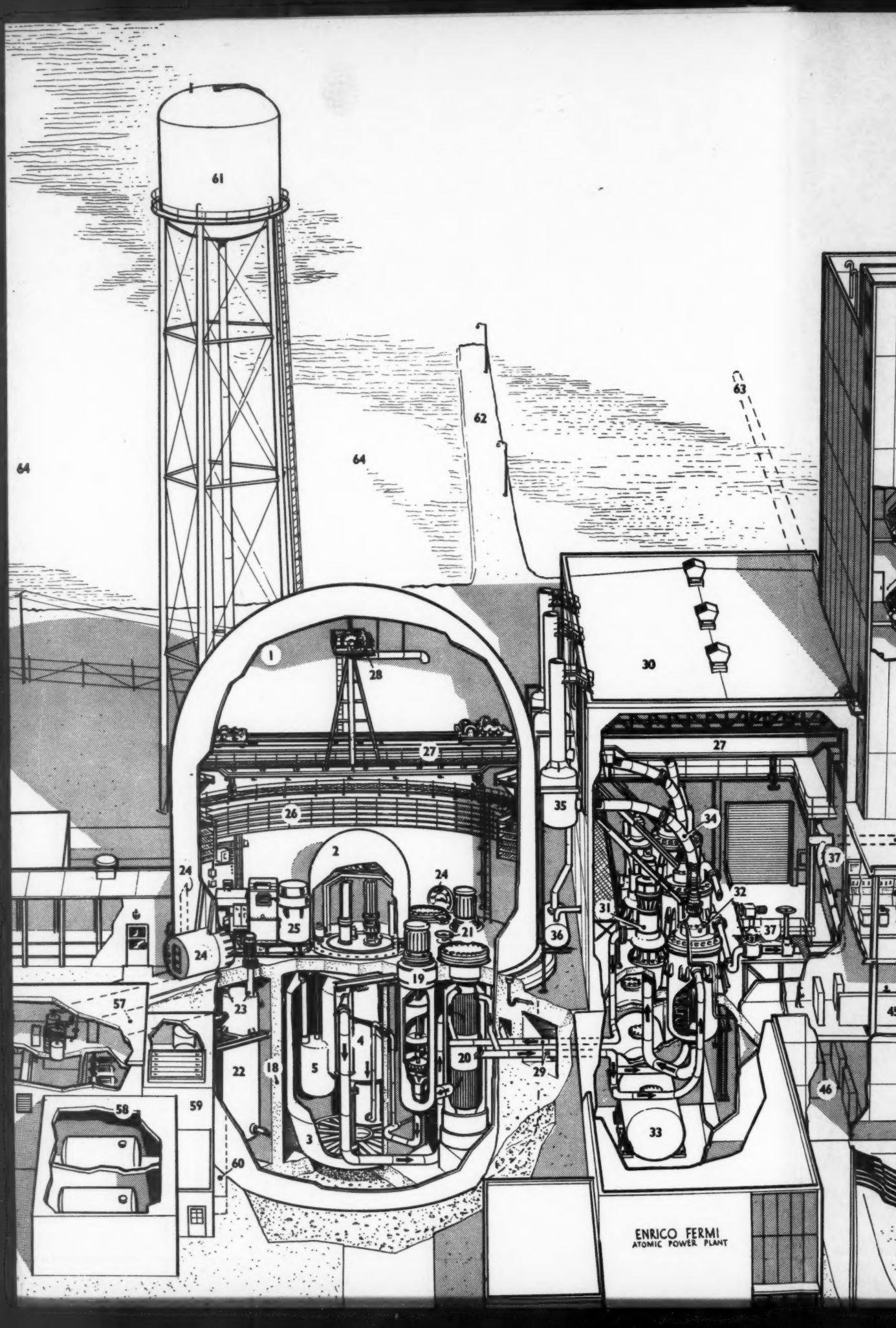
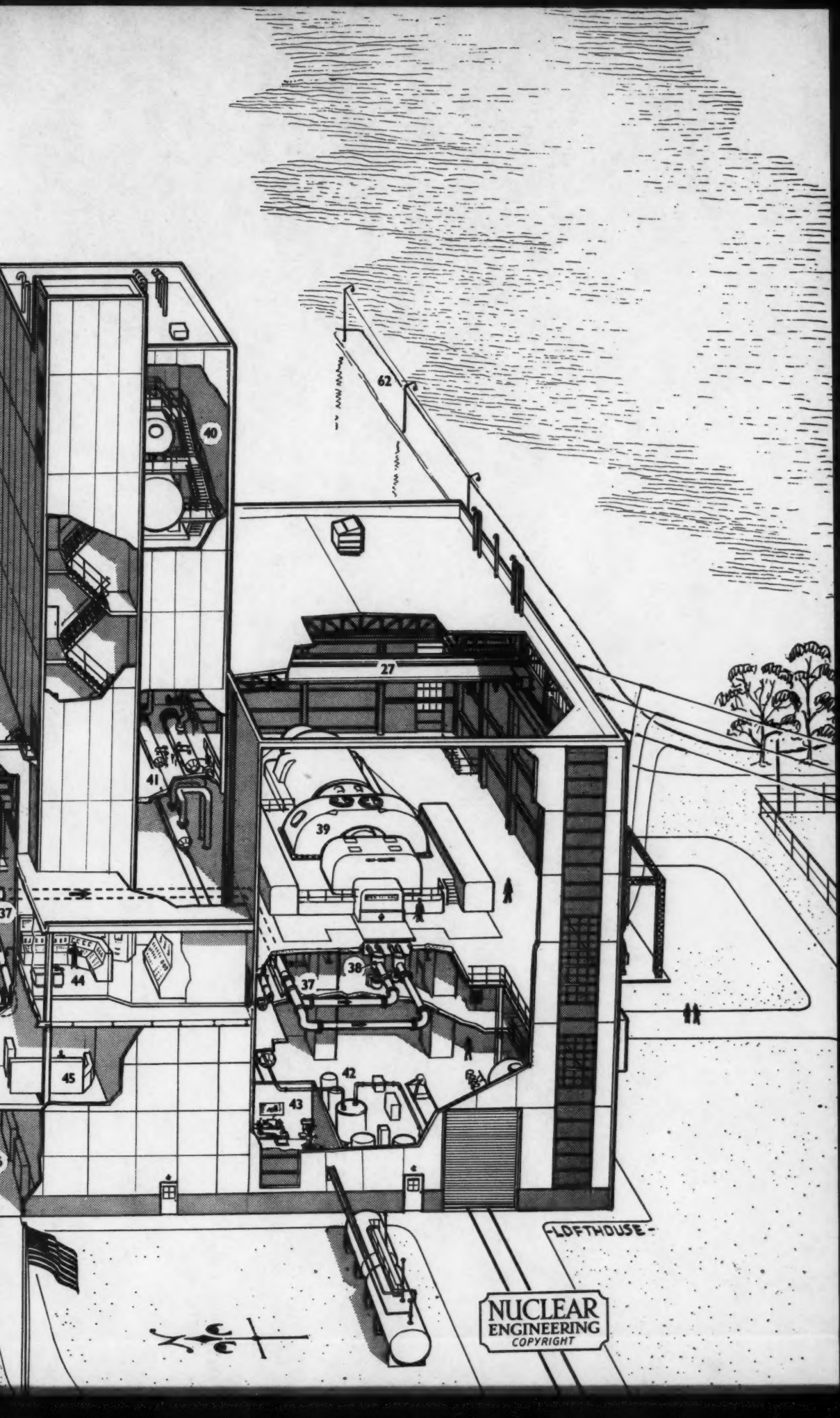


Fig. 4.—(Above) Simplified section of reactor.

Enrico Fermi




ENRICO FERMI
ATOMIC POWER PLANT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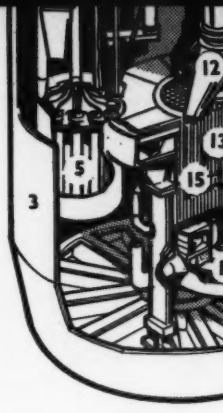


The

No

PLANT D
WIT

OWNER:

DESIGN:

LOCATION:

TYPE:

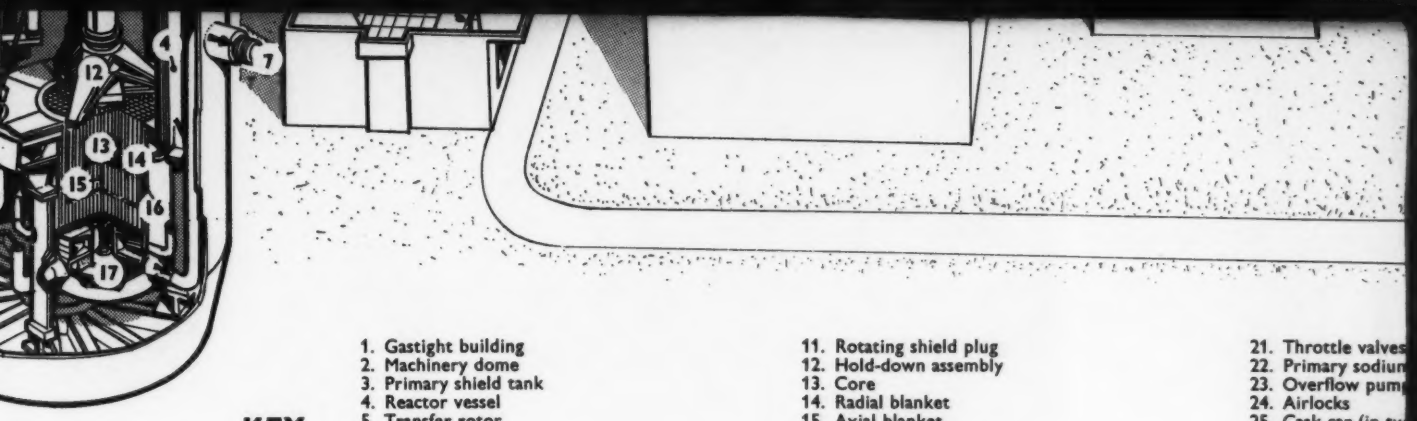
PURPOSE:

PROGRAMME:

CAPACITY:

FISSILE:

FERTILE:



KEY

1. Gastight building
2. Machinery dome
3. Primary shield tank
4. Reactor vessel
5. Transfer rotor
6. Coolant inlet
7. Coolant outlet
8. Control rod mechanism
9. Offset handling mechanism
10. Transfer tube
11. Rotating shield plug
12. Hold-down assembly
13. Core
14. Radial blanket
15. Axial blanket
16. Thermal shield
17. Melt-down pan
18. Secondary shield wall
19. Primary sodium pumps
20. Intermediate heat exchangers
21. Throttle valves
22. Primary sodiu
23. Overflow pum
24. Airlocks
25. Cask car (in tw
26. Cable galleries
27. Overhead cran
28. Atmosphere co
29. Secondary sodi
30. Steam generati

The World's Reactors

No. 33 ENRICO FERMI

PLANT DESIGN AND PERFORMANCE DATA WITH INITIAL U-Mo ALLOY CORE

:	Power Reactor Development Company.
	Atomic Power Development Associates, Inc.
ON:	Lagoona Beach, (near Monroe) Michigan, U.S.A.
	Sodium cooled fast breeder reactor.
E:	Power generation; breeding.
AMME:	Site work commenced: April, 1956.
TY:	Thermal output: 200 MW. Electric output: 65·9 MW gross. 60·9 MW net. Conversion ratio: 1·16.
	Enriched U-10 w/o Mo alloy. Enrichment: 25·6 w/o. Total loading: 4 270 lb. Critical mass (105 sub-assemblies): 496 kg U^{235}. Core conversion factor: 0·29.
:	Depleted U-3 w/o Mo alloy. U^{235} concentration: 0·35 w/o. Total loading: 67 780 lb. Blanket conversion factor: 0·87.

FUEL ELEMENTS:	Fuel pins: 0·158 in o.d. × 32·78 in total length. Uranium portion of pin: 0·148 in dia. × 30·5 in. Cladding: 0·005 in Zr. Pin pitch: 0·199 in, square. No. of pins per sub-assembly: 140. No. of sub-assemblies in core: 105. Sub-assembly dimensions: 2·646 in square × 97·489 in long; wall thickness: 0·096 in; "active" length of sub-assembly = 17 in (blanket) + 1 31/64 in (space) + 32 25/32 in (core) + 2 55/64 in (space) + 17 in (blanket).
BLANKET ELEMENTS:	Breeding rods: 0·443 in o.d. × 65 in (radial). 0·415 in dia. × 14 in (axial). Cladding: 0·010 in s.s. type 304. Sodium bond: 0·014 in. Rod pitch: 0·483 in (radial and axial), square. No. of rods per sub-assembly: 25 (radial). No. of rods per section: 16 (axial). No. of sub-assemblies: 531 (radial). No. of sections: 105 (in each axial region).
CORE:	Dimensions: 32·7 in dia. × 30·5 in (active length). Active volume: 13.4 ft^3. Power: 174 MW. Heat transfer surface: 1 545 ft^2. Maximum heat flux: 640 000 Btu/ft^2h. Power density: 13 MW/ft^3. Specific power: 0·35 MW/kg U^{235}. Maximum local removal burn-up: 0·4 a/o. Max. nom. U temp.: 1 115° F. Max. nom. cladding surface temp.: 1 051° F.
BLANKET:	Radial dim.: 30·5 in i.d. × 79·9 in o.d. × 65 in total length; length of U alloy: 61·75 in. Axial dim.: 30·5 in dia. × 14 in active length (each region). Active volume: radial: 138 ft^3; axial: 6·1 ft^3 (per region). Power: radial: 23·6 MW; axial: 1·2 MW (per region). Heat transfer surface: radial: 7 923 ft^2; axial: 227·5 ft^2 (each). Maximum heat flux: radial 217 000 Btu/ft^2h axial 72 200 Btu/ft^2h.

PRIMAR
COOL

REACT
PHYS

21. Throttle valves
22. Primary sodium overflow tank
23. Overflow pumps
24. Airlocks
25. Cask car (in two positions)
26. Cable galleries
27. Overhead cranes
28. Atmosphere conditioning unit
29. Secondary sodium piping
30. Steam generating building
31. Secondary sodium pumps
32. Steam generators
33. Sodium storage tanks
34. Sodium-water reaction vents
35. Sodium separator units
36. Feedwater dump tank
37. Main steam line
38. Main steam stop valves
39. Turbo-generator
40. Storage and de-aerators
41. L.P. heaters
42. Water treatment plant
43. Workshop
44. Main control room
45. Reactor simulator
46. Switch room
47. Covered car track
48. Fuel handling building
49. Repair pit
50. Transfer tank rotor

in total length.
n dia. × 30·5 in.

40.
105.
46 in square × 97·489 in
in; "active" length of
t) + 1 31/64 in (space) +
n (space) + 17 in (blanket).

5 in (radial).
4 in (axial).

axial), square.
25 (radial).
ial).
ial).
ial region).

in (active length).

u/ft²h.

p: 0·4 a/o.

np.: 1 051° F.

in o.d. × 65 in total length;

ctive length (each region).

er region).

ion).
923 ft²;
7·5 ft² (each).
000 Btu/ft²h
0 Btu/ft²h.

PRIMARY COOLANT:

Sodium.
Inlet temp.: 550° F.
Exit temp.: 800° F.
Max. nominal coolant temp. in core: 1 014° F.
Flow through core: $7{\cdot}10 \times 10^6$ lb/h.
Flow through radial blanket: $1{\cdot}47 \times 10^6$ lb/h.
Velocity in core: 15·7 ft/s.
Total Na volume: 7 260 ft³ (421 000 lb at 58 lb/ft³).
No. of primary circuits: 3.
Pumps: 3 motor driven vertical shaft centrifugal sump type.
Pump capacity: each, 11 800 gal/min at 310 ft.
Drive: 1 000 h.p., 900 rev/min wound rotor with liquid rheostat control.
Piping: 30 in flow, 16 in and 14 in return, all $\frac{3}{8}$ in wall, in type 304 s.s.

REACTOR PHYSICS:

Critical mass: (105 sub-assemblies) 496 kg U^{235}.
Median flux energy: 0·32 MeV.
Mean fission energy 0·25 MeV.
Average core flux: 3×10^{15} n/cm² s.
Average neutron generation time: 0·084 s.
Mean delay time of delayed neutrons: 12 s.
Prompt neutron lifetime: $1{\cdot}4 \times 10^{-7}$ s.
sothermal temp. reactivity coefficients, 10^{-6} $\Delta k/k$ °C :

	Core region	Axial blanket region	Radial blanket region
Sodium expansion	−4·4	−5·2	−1·7
Uranium alloy expansion	−6·2	−0·1	−0·2
Hold-down plate expansion	0·0	−0·4	−0·3
Doppler (at 550° F)	−2·6		
Sub-assembly can radial expansion	−11·3		−0·7
Lower support expansion	−1·5		
Power Coefficients, 10^{-3} ¢/MW			
Sodium expansion	−2·1	−2·7	−0·9
Uranium alloy expansion	−6·8	−0·1	−0·2
Hold-down plate expansion	0·0	−0·2	−0·1
Bowing	+0·2		
Doppler (at 200 MW)	−1·9		
Sub-assembly can radial expansion	−9·4		−0·3

Reactivity requirements, weekly unloading	¢	% $\Delta k/k$
Temp. override	60·0	0·40
Total loss, per week	15·4	0·10
Control margin	16·6	0·11
	92·0	0·61

CONTROL:

Effective delayed fraction (β):
Safety rods: 8.
Total controlled: 0·061 Δk
0·0066 $\Delta k/k$.
Rods comprise: 6 tubes of
to 59 w/o in B^{10} clad in s
inner cladding 0·01 in, ou
Total B^{10} weight per rod: 5
B_4C tube dimensions: 36·3
outer 0·625 in o.d.
Rod dimension $2\frac{1}{4}$ in o.d. ×
Scram shut down average r
scram: 0·11 $\Delta k/k$ s.
Operating control rods: 2
$\Delta k/k$.
Rods comprise: 19 cylinder
in B^{10} clad in 0·028 in of
B_4C cylinder dimensions:
length 30 in.
Rod dimensions: $2\frac{1}{4}$ o.d. ×

REACTOR VESSEL:

S.s. type 304, 14·5 ft max. di
Wall: 2 in max.
High pressure plenum, 110 p
Radial blanket plenum, 50 ps
Upper vessel, 50 psi, 1 000° F

INTERMEDIATE H.E.s:

3, counter flow shell and tub
No. of tubes per shell: 1 860.
Tube size: $\frac{5}{8}$ in o.d. × 0·049 in
Heat transfer per H.E.: 489 ×
Heat transfer surface: 4 840 f

SECONDARY COOLANT:

Secondary coolant: Na.
Secondary flow (in tubes):=
lb/h per H.E.
Temp. rise: 517° F–767° F.
Total secondary Na wt.: 2
(at 58 lb/ft³).
Piping: 18 and 12 in with $\frac{3}{8}$ in
Secondary pumps, double-v
mounted.
Capacity: 12 000 gal/min at 3
Drive: 350 h.p., 900 rev/min
current couplings.

51. Cleaning chamber and equipment
52. Cut-up pool
53. Decay pool
54. Health physics laboratory
55. Sodium service building
56. Sodium control room
57. Sodium tunnel
58. Waste gas building and decay tanks
59. Inert gas building
60. Inert gas tunnel
61. Potable water storage tank
62. Piers
63. Water intake channel
64. Lake Erie
65. Stack

fraction (β): 0·662% $\Delta k/k$.

ed: 0·061 $\Delta k/k$; each controlling at least

: 6 tubes of B_4C " doughnuts " enriched
B^{10} clad in stainless steel type 316 tubes;
ng 0·01 in, outer 0·032 in.
ht per rod: 535 g.
ensions: 36·3 in long; inner 0·312 in o.d.;
in o.d.
n 2½ in o.d. × 68⅛ in.
wn average rate of reactivity insertion on
$\Delta k/k$ s.
rol rods: 2, each controlling 0·305%

e: 19 cylinders of B_4C enriched to 34 w/o
n 0·028 in of s.s. type 316.
dimensions: 0·254 in dia. × 10 in; tube
.
ns: 2½ o.d. × 42 17/64 in.

·5 ft max. dia. × 36·3 ft high.

lenum, 110 psi, 750° F.
lenum, 50 psi, 1 000° F.
psi, 1 000° F.

shell and tube type in type 304 s.s.
shell: 1 860.
o.d. × 0·049 in wall.
r H.E.: 489 × 10⁶ Btu/h.
rface: 4 840 ft².

nt: Na.
(in tubes):= primary flow= 2·95 × 10⁶

° F–767° F.
Na wt.: 256 300 lb; volume 4 418 ft³

2 in with ⅜ in wall in 2¼% Cr–1% Mo steel.
ps, double-volute centrifugal, vertically

gal/min at 35 psi.
900 rev/min induction motors with eddy
gs.

STEAM GENERATORS:
3 once through cross and counterflow.
Material: 2¼% Cr–1% Mo steel.
Steaming capacity per unit: 211 560 lb/h.
Steam conditions: 600 psia, 764°F.
Feedwater temp.: 340° F.
Heat transfer areas: economizer: 965 ft².
evaporator: 2 165 ft².
superheat: 7 670 ft².

SHIELDING:
Primary radial: blanket, 12 in s.s., 6 in of 5% borated graphite, layer of insulation, 2½ ft graphite, 6 in of 1% borated graphite.
Secondary radial: 2½ ft concrete with ½ in steel liners+ 2 in steel baffle opp. Na loops.
Biological shield: 7 ft concrete outside 1·03 in reactor building.
Top shield: s.s. shell containing from bottom to top: six 11-in layers of plain graphite, one 11-in layer of 1½% borated graphite, 12 in carbon steel, 1 ft s.s. insulating material, 11 in cover plate of solid carbon steel.
Operating floor: steel-and-concrete 5 ft thick.

REACTOR BUILDING:
Cylindrical steel vessel 72 ft i.d. with hemispherical top, ellipsoidal bottom.
Height: 120 ft, 51 ft below grade.
Material: ASTM A–201 grade B.
Design pressure: 32 psig.
Design temp.: 550° F.

GENERATING PLANT:
Tandem compound single flow machine.
1 800 rev/min.
Gross capability: 150 MW, 1 inHg back pressure.
Operating performance at reactor power of 200 MW (th):
Gross output: 65·9 MW; net output 60·9 MW.
Steam conditions (at turbine): 576 psia, 760° F.

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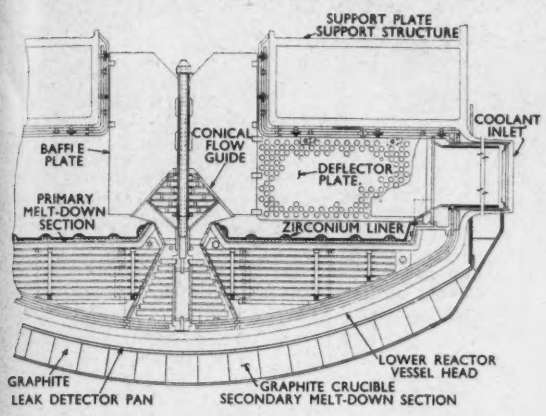
Fig. 6.—Section of bottom of reactor vessel, showing melt-down arrangements.



for the core coolant inlet pipes; the three 6 in nozzles for the blanket coolant inlets are 4 ft from the bottom, so that they open directly into the annular structure which forms the blanket plenum. The central hole, forming the core inlet plenum, is fitted with conical flow guides and baffles.

This conical flow guide, as well as directing coolant flow in normal operation, is also designed, in case of a melt-down, to disperse the molten fuel and prevent a super-critical mass forming in the centre. A zirconium liner, $\frac{1}{8}$ in thick, forms a tray or pot to catch any molten metal and, in the case of a complete melt-down, the fuel would be contained as a slab 7 ft diameter and 1·25 in thick. This tray is supported on the layers of stainless steel baffles and thermal shields beneath. Should the zirconium be perforated by the molten fuel, secondary containment is provided below the vessel bottom by an assembly of 6 in borated graphite cubes bonded together by a boron-containing cement and forming a crucible which would contain molten uranium.

A transition section, consisting of a flat plate with welded stiffening ribs, serves to connect the lower vessel with the

Fig. 7.—Building up shielding inside the rotating plug.

considerably larger (14 ft 2½ in i.d.) upper vessel and the transfer rotor container. This transition section also carries the vessel support brackets. The supports take the form of 2 in steel plates 7 ft long to give flexibility; they are arranged so that during expansion the centre line of the upper vessel remains fixed.

Shield Plug

The top portion of the upper vessel narrows down to accommodate the rotating plug, which serves the dual purpose

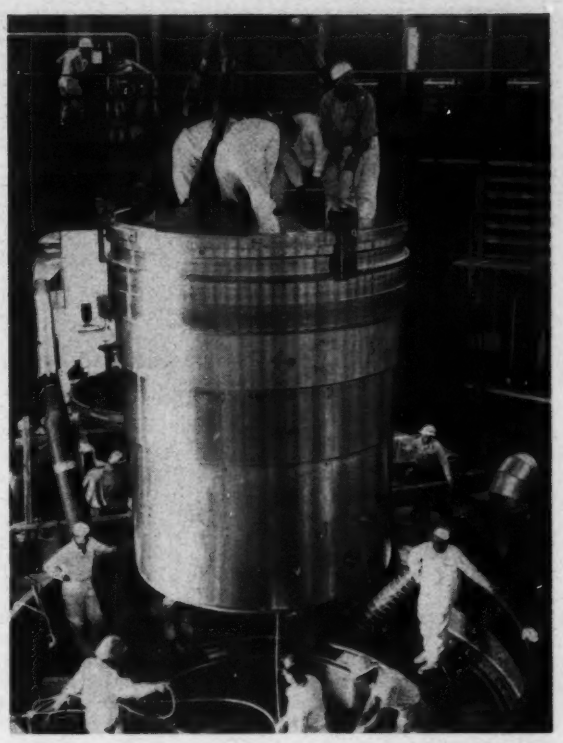
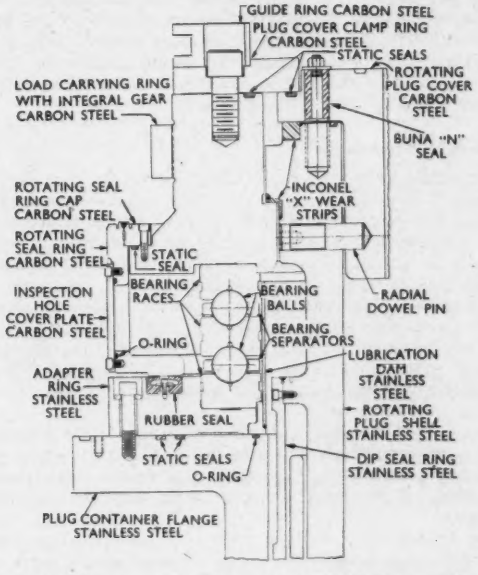


Fig. 8.—(Above) Shield plug being lowered into position.

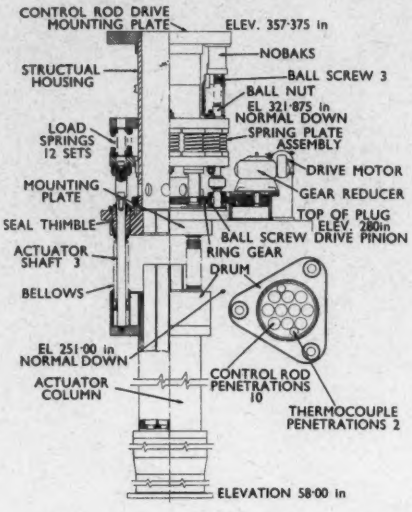
Fig. 9.—(Below) Bearing and sealing arrangements for the shield plug.



of top shielding, and location of the hold-down and fuel-handling mechanisms. It consists of a stainless steel shell approximately 12 ft 3 in high, of diameters stepped from 9 ft 4 in to 8 ft 10 in, and is $1\frac{1}{32}$ in thick, and is filled with insulating and shielding materials in layers, including 18 in of stainless steel, 6 ft of graphite (of which some 20% is borated to 1·5 a/o) and 2 ft of carbon steel. The top plate is of carbon steel, 12 in thick, to ensure that the deflection is limited to 0·015 in, so that the handling mechanism shaft will remain within an angle of 0·5′ of the vertical.

The weight of the plug (120 ton) is carried by a carbon steel

Fig. 10.—(Left) Part of hold-down mechanism during erection.
Fig. 11.—(Below) Simplified section of hold-down mechanism.



load-carrying ring, with integral gear teeth, and transmitted to the flange of the plug-containment portion of the vessel through a double-row ball bearing, the intermediate race ring between the two rows of balls allowing for differential expansion of the stainless steel flange and the carbon steel load-carrying ring. Molybdenum disulphide is used for lubrication.

Sealing between the plug and the vessel is obtained by a mechanical seal; provision is made for a NaK dip seal if this should be required. All bolted joints and covers are fitted with O-rings.

Large-diameter stepped holes are provided for the hold-down and handling mechanism columns. In addition, a 3 in hole is provided in the centre of the plug, for lining-up during erection, and three 6-in holes providing access for removing the control rod lower guide tubes and handling the beryllium section of the neutron source. These holes are filled with shield plugs during operation.

Above the plug and enclosing the mechanism is the machinery dome, a removable structure which is, in effect the upper head of the primary shield tank, and it has a twofold function—to act as gas containment in case of the plug seal leaking, and as missile containment in case of a violent explosion. The top is fitted with 6 ft of honeycomb aluminium material with a load equalizing plate; by crushing, this would absorb the impact stresses caused by the plug being ejected (with a force sufficient to raise the plug 170 ft if unrestrained).

Hold-down Mechanism

The actual hold-down head, which locates the core elements to prevent inward movement and rising under the force of the upward flow of coolant, takes the form of a three-armed spider, each arm being fitted with a socket which fits over a support column for accurate location, carrying a cluster of hold-down fingers each of which fits the handling head of the appropriate core element. The head is attached to the hold-down column, which passes through the shield plug. The hold-down column does not rotate in the plug, but is capable of being raised and lowered some 9 in, to clear the core and allow the shield plug to rotate and position the offset handling mechanism over the core.

Raising and lowering, and the application of the hold-down force (approximately 50 ton) is by three ball screws synchronized by a ring gear, motor-driven through a double worm reducer, and three bellows-sealed actuator shafts through compression springs.

The hold-down column is hollow, to accommodate control and safety rod drives, there being 10 penetrations for control rods and two for thermocouple leads.

Offset Handling Mechanism

The handling mechanism consists, in essence, of a column penetrating the shield plug, capable of rotation (to an accuracy of 0·1° of azimuth) so that, by combined column and shield plug rotation, the offset arm at the lower end, which carries the gripper fingers, is capable of location over any core or blanket position.

The gripper arm is mounted on a tube telescoping into the main column, and is capable of being raised and lowered sufficiently for the core elements to clear the top of the core, and is steadied for vertical positioning by a triangular braced structure carried on the main column. Dual-speed operation is provided for lifting and lowering; the normal speed of 20 ft/min is automatically slowed to 1 ft/min as the fuel portion of an element enters the core and, should there be any sign of the reactor approaching criticality, the mechanism is automatically reversed and the element withdrawn at full speed.

The gripper fingers are locked in position by means of a probe which first contacts the element head and checks the downward movement when the arm has reached the correct position for gripping. It is not possible for the gripper fingers to be inadvertently unlatched while carrying an element; it is necessary for the element to be seated at the correct elevation before the solenoid-operated lock can be released.

Since the fuel elements are square, it is necessary that they should be correctly oriented to be placed smoothly in position either in the core or in the transfer rotor. This is accomplished by providing a simple cam at the top and bottom of each element which provides self-orientation; this feature

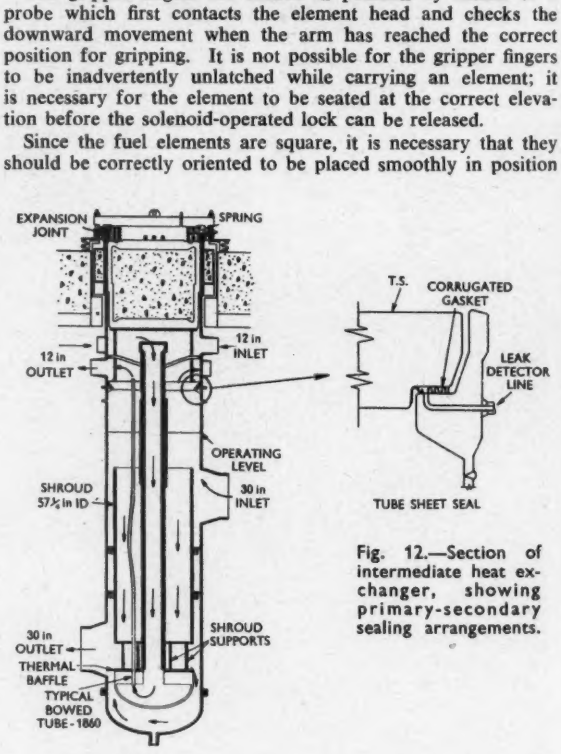


Fig. 12.—Section of intermediate heat exchanger, showing primary-secondary sealing arrangements.

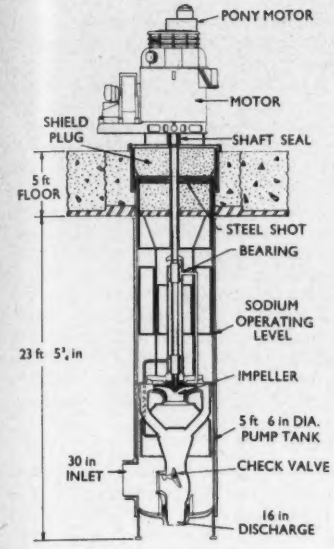
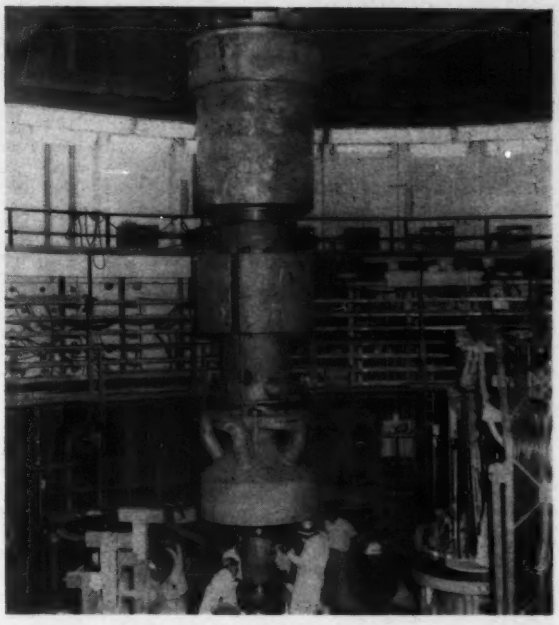


Fig. 13.—(Left) Section of primary sodium pump as installed.

Fig. 14.—(Right) Pump and shield plug, showing unit construction.



avoids the additional complication of rotating mechanisms for the gripper head.

Some of the azimuth positions of both plug and handling mechanism for location of various fuel elements were determined manually, the remainder by computer. Check tests at room temperature and in hot air showed that it was possible to pick up any element at the computed positions.

Transfer Rotor

The transfer rotor consists of a rotating disc with driveshaft and indexing mechanism, providing temporary storage for 11 core assemblies in sodium-filled finned pots. Provision is made for sealed withdrawal of the pots or insertion of new elements, through an exit port immediately above the rotor; fuelling arrangements will be dealt with rather more fully at a later stage.

Primary Coolant Circuit

Brief reference has already been made to the dual flow of sodium through the core and blanket regions of the reactor. Nearly 90% of the total flow passes through the core, and the two flows meet in a common pool towards the top of the reactor vessel, from where 30-in pipes feed the three coolant loops. Flow from the vessel through each loop is by gravity through the shell side of the intermediate (sodium/sodium) heat exchanger, then to the pump and to the reactor inlets, the 6-in inlet to the blanket plenum being tapped off the main core flow not far from the pump outlet, by means of a throttle valve allowing adjustment of the blanket flow.

Fig. 15.—Secondary (tube unit) of IHX being lowered into shell.

The intermediate heat exchangers are vertical shell-and-tube type, suspended from the floor above, and fitted with a shielded plug to enable access to be had to the secondary tubes without draining or dismantling the primary circuit. The inlet and outlet for the primary sodium are on the side of the shell.

The secondary sodium enters at the top and flows through a central downcomer to the floating head, whence it returns to the secondary outlet through 1 860 bowed tubes each $\frac{7}{8}$ in o.d. Stainless steel shielding around the downcomer protects it from thermal shock. A piston ring seal is used in the divider plate separating incoming and outgoing secondary sodium; some leakage here is accepted as unimportant. Leakage between the primary and secondary sodium is prevented by a corrugated nickel and asbestos gasket between the upper tube plate and the shell, compressed by Inconel X springs acting through the shield plug; leak detectors are fitted. The total heat transfer surface of each heat exchanger is 4 840 ft², and the heat transfer 489 × 10^6 Btu/h.

Each pump is of the vertical-shaft centrifugal type, driven by a 1 000 h.p. wound rotor induction motor with liquid resistance speed control, the normal full load speed being 860 rev/min, and the pump is designed to have a capacity of 11 800 gal/min (U.S.) at a total dynamic head of 310 ft. To prevent escape of inert gas from the pump shaft, a rotating seal is provided, lubricated by a fluorocarbon oil containing no hydrogen.

The entire pump is enclosed in a steel tank 5 ft 6 in diameter, which extends from below discharge level right up to operating floor level; this enables the entire shield plug, shaft, impeller and casing to be removed without draining the primary system. A check valve, to prevent back flow in case of pump failure, is incorporated. The primary pumps remain in operation after all scrams except those which are initiated by complete failure of electrical supplies to the pump motors. Failure of one pump will enable operation to continue at reduced output; failure of two will cause a scram. In case of complete failure, the natural circulation obtained in the primary circuits will be assisted by pony motors on the pumps, fed from guaranteed supplies to remove decay heat.

The 6-in throttle valves controlling the blanket plenum are of the right-angle pattern, double-bellows sealed, and the entire valve assembly is contained within a pipe riser sealed above sodium level so that it can be withdrawn without draining the primary circuit. Flow can be regulated from 5% to 20% of the total, but not shut off altogether.

Although most of the primary system components have been designed to permit removal without draining the sodium, a drain system is provided to the IHX shells, the drain lines leading to the overflow tanks, and the sodium being allowed to "freeze" a few feet from the IHX.

Loss of primary sodium is, probably, the most serious fault which could occur. To give decay cooling under all conditions, it is essential that the level should not drop below the 30-in reactor outlets. In the event of a leak occurring in the vessel portion, the secondary shield tank would assist in this, but the spaces within the tank would eventually fill. Reserves available include the excess level in the reactor vessel itself, the overflow tanks, and reserve sodium in the sodium service building.

Additional steps taken to ensure against leakage of the primary sodium include enclosure of the gear by "double-skin" construction. For example, the 30-in piping is surrounded by a 36 in Cr-Mo enclosure, $\frac{1}{16}$ in thick made in two halves and welded into position after completion and testing of the primary circuit. The IHX, pump tanks, discharge piping, etc., are surrounded by $\frac{3}{4}$ in thick carbon steel, allowing a $2\frac{1}{2}$ in annulus between the containment and the primary circuit components.

The sodium surfaces in the reactor and in the pumps are covered by an argon blanket. Syphon breakers, to prevent the sodium in the reactor vessel being syphoned out in the event of a fault on the pressure side of the pumps, are provided, and a small flow of sodium is continually maintained to ensure that they are clear in case of need.

All piping upstream of the IHX will be cold sprung for operation at 900°F; on the pressure side of the pump cold-springing is carried out for 600°F. Thermal shock studies and vibrational analyses have been made on the system.

The total volume of the sodium in the primary circuit is some 5 450 ft^3 (345 200 lb) and expansion as the reactor approaches working temperature is dealt with by the over-flow tanks and pumps within the reactor building. First filling and subsequent purification is carried out in the sodium service building (immediately to the left of the reactor in the pull-out drawing) which is connected to the reactor by a shielded tunnel and contains sodium tanks and the necessary cold traps and associated heat exchangers for oxide removal, with plugging indicators for assessment of oxide content. Associated with the sodium service building is the inert gas handling plant.

Throughout the plant, heating has been provided to allow of filling and to prevent freezing even in the case of a prolonged shut-down. Wherever possible, 120 V induction heating has been applied—in most cases outside the pipe insulation. In certain locations stainless steel pipes have been clad with carbon steel to make this inductive heating practicable. Other components, including the reactor vessel, have been fitted with resistance heaters. Throughout, careful sequential control has been arranged for heating circuits to eliminate the possibility of pipe rupture through thawing a trapped section of sodium.

CONTROL

In considering the control of the Enrico Fermi plant, it must be borne in mind that many of the characteristics of a fast reactor are very different from those of an equivalent thermal reactor—for example, the excess reactivity required to overcome Xe and Sm poisoning is negligible and temperature coefficients are about an order of magnitude lower. It was decided early in the design, to limit the excess reactivity in the core to less than 1%, and the actual final figure is considerably less than this.

The design provides for two control rods each worth $0{\cdot}31\%\Delta k/k$ and eight safety rods, each worth $0{\cdot}75\%\Delta k/k$. Only the control rods are normally within the core; safety rods are normally fully withdrawn, and are only introduced for scram purposes and to maintain the reactor sub-critical during refuelling.

The drive extensions for both control and safety rods are accommodated in the hollow column of the hold-down device, in the rotating shield plug above the reactor. The control rods are arranged almost at the centre of the core, the safety rods surrounding them in an elliptical ring.

Accommodation within the core is provided, for both safety

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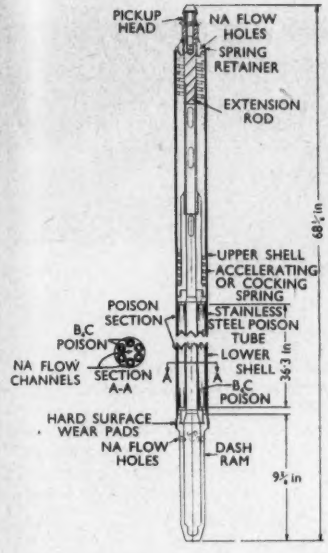
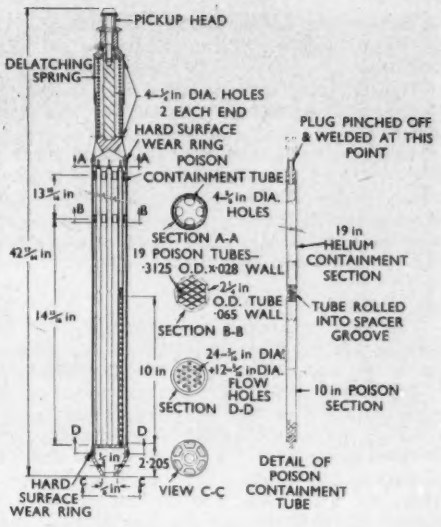


Fig. 16.—(Left) Section of safety rod, showing arrangement of self-contained accelerating spring.

Fig. 17.—(Right) Section of operating control rod, showing details of poison section.



and control rods, by guide tubes. These have a circular section to guide the rods, and are contained in a square can the same size as a fuel element. The safety rod guides are fitted with a dashpot, the control rod guides are not. Both types have breakdown orifices at the lower end, to restrict the coolant velocity to a value which does not tend to lift the rods, or impede their downward movement.

All rods are fitted with pick-up heads similar to those of the fuel elements, so that they can be delatched from their operating mechanisms; this must take place before the hold-down device can be raised and the plug rotated, for refuelling.

The safety rods may be raised slowly, driven in fast, or scrammed. Raising, for start-up, or slow lowering, takes place at a speed of 1·6 in/min, equivalent to 0·01% reactivity per second for the eight rods moving simultaneously. Automatic stops of 2 min duration occur during withdrawal at sub-critical levels of 0·5% and 0·25%. Fast drive into the core occurs at 10 ft/min. Scramming is by simply delatching the rods and allowing them to be driven in by springs; the spring is built into the rod itself and is compressed as the first movement of the latch-on sequence, so that scram can take place even if the rods are only partially raised.

The control rods are operated as a shim rod, with a fixed speed of 0·4 in/min in either direction and a regulating rod, with a variable speed of 1-10 in/min as required.

Boron carbide is used as the poison in both control and safety rods and, although the two types differ sharply in design, in both cases the containment for the poison is designed as a miniature pressure vessel, in order to contain the helium generated by neutron irradiation. The safety rods, which are right out of the core during normal operation are estimated to generate 42 litres of free helium in a year; the control rods 40 litres a year, and some 10% of the gas in each case would be released from the lattice to exert pressure.

Control Rods

The poison section of the control rods is about 30 in long, and consists of 19 hermetically sealed stainless-steel tubes, arranged on a triangular pitch inside a stainless-steel tube 2½ in o.d. The upper 19 in is left as a void section for gas expansion, the lower 10-in section is filled with compressed boron carbide in the form of ¼-in cylinders; they are restrained by a plug with a small central hole; the plug is located by rolling the tube into a circumferential groove.

Safety Rods

The poison section of a safety rod is 5 in longer than the core and consists of six annular rods (hollow cylinders) of hot-pressed boron carbide canned in stainless steel, with a stainless-steel centre tube to provide accommodation for the helium. The six rods are contained in a stainless-steel shell through which the coolant flows.

Below the poison section is the ram, a doubly tapered piston, 6 in effective length, which enters the dashpot at the bottom of the guide tube when scram takes place and gives a deceleration of about 3*g*, from the maximum velocity of 6·5 ft/s.

Above the poison section is the extension rod ending in the pick-up head; this section also contains the accelerating spring which must be compressed before the rod is picked up.

When it is desired to remove the complete safety rod sub-assembly for replacement, it cannot be picked up by the offset handling mechanism without first removing the four adjacent core elements, thus eliminating the possibility of accidental criticality.

Oscillator Rod

This device, while not forming a permanent feature of operation, will be used for experimental work in the early stages. The oscillator is a boron carbide cartridge, eccentrically mounted and capable of being rotated so as to give small cyclical variations in reactivity (about 0·03% peak-to-peak) at any speed from 0·06 rev/min to 600 rev/min, by variable speed motor drive. The cartridge is mounted in one of the positions normally occupied by a safety rod.

General Control Philosophy

The reactor power level will determine load. The key point of this philosophy is the reactor outlet temperature, and setting of a power requirement on the automatic system control is, in effect a demand for a particular temperature, as the flow rates for both primary and secondary coolant remain constant. Reactor power level changes are subject to a limit (built into the automatic gear, and independent of the operator) of 3 MW/min. The demanded rate of change of power is compared with the actual rate, determined from the flux count, and the error is used to drive the selected control rod in the appropriate direction.

The steam side will follow the reactor output and automatic control of turbine throttle and feedwater flow will keep constant steam pressure, any excess steam being dumped.

STEAM PLANT

Consideration of the steam plant really includes all apparatus outside the reactor building, and begins with the secondary sodium loop from the intermediate heat exchangers.

Each of the three secondary sodium loops has a flow of $2{\cdot}95 \times 10^6$ lb/h at 200 MW. From the twin inlets and outlets of the IHX, 12 in pipes are taken through the wall and

shielding of the reactor building and branched into 18 in pipes in the E. and W. sodium galleries to the steam generator building.

The secondary sodium pumps, like the primary pumps, are vertical-shaft centrifugal type, but the shafts are, of course, considerably shorter. Each pump has a design capacity of 13 000 gal/min at a total dynamic head of 100 ft, and is driven by a 350 h.p., 900 rev/min motor, speed variation being by means of an eddy-current coupling. Pony motors are also provided on the secondary pumps to operate, in conjunction with the primary pump pony motors, in decay heat removal on loss of main pump supplies.

Steam Generators

The steam generators are vertical shell-and-tube type, the sodium flow being in the shell, with two inlets near the top of the unit and a single outlet at the bottom.

The steam side is a once-through cross-counterflow arrangement of U-tube bundles, water entering at the top where there is a ring header, and passing down through vertical tubes in a shrouded area in the centre of the shell and upwards through horizontal loops to a ring steam header near the top. There are 1 200 tubes, $\frac{5}{8}$ in dia. o.d., giving a total heating surface of 10 800 ft^2 for the feed-heating, evaporator and superheater sections. Steam at 600 psia, superheated to 764°F, is generated.

In order to protect the system in case of a water/sodium leak, each steam generator is fitted with a relief diaphragm or bursting disc, allowing gaseous products to be vented outside the building, where centrifugal separators are provided to prevent ejection of particulate matter.

Turbine

The turbo-generator set is designed so that the station rating can, if necessary, be increased, and has an ultimate gross capacity of 156 MW (corresponding to a reactor output of 430 MW(th)). Initially, the steam conditions will be 576 psi and 760°F; ultimate steam conditions will be 865 psi and 780°F.

The turbine is a single-flow, tandem-compound unit running at 1 800 rev/min; this speed was selected to make use of standard last-row blades for the l.p. cylinder. This size (46 in) is, incidentally, the largest currently in use.

In order to keep the moisture content of the exhaust end within acceptable limits, inter-stage reheating is provided between the h.p. and l.p. cylinders. The twin reheaters are located at the sides of the turbine and take the place of the normal cross-over connections between the cylinders, resulting in a very neat and compact design.

Fig. 19.—General view of steam generator house.

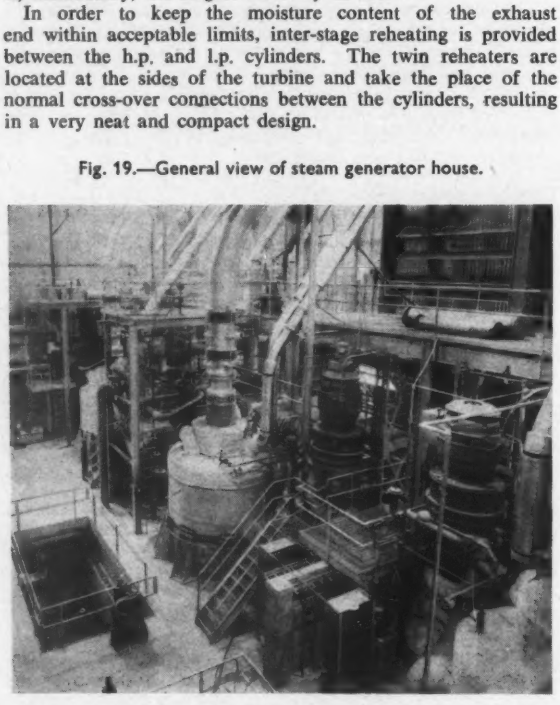


Fig. 18.—The Allis-Chalmers turbo-generator. Note steam reheaters located in side cross-overs.

The l.p. cylinder is, basically, one-half of the standard design used for modern 275-400 MW units. No modern design, however, was adaptable for the rather low steam conditions of the h.p. cylinder, and an entirely new design was used, straight reaction blading being employed throughout.

Three-stage feed heating is provided with a fourth (deaerating) heater of large size. When steam dumping is necessary, steam is released into this and, via a second dump valve, into the condenser, through break-down orifices and a desuperheater. The condenser can accept steam dumping up to 750 000 lb/h; any excess steam is released to atmosphere via the safety valves on No. 4 heater.

CONTAINMENT AND SHIELDING

As can be seen from the pull-out drawing, the reactor vessel, etc., are immediately surrounded by the primary shield tank, a $\frac{5}{8}$-in thick carbon steel vessel 24 ft maximum diameter and nearly 40 ft high, which extends through the operating floor. The top portion, including the hold-down device and its associated control mechanisms, is enclosed within a machinery dome, a double-skin unit partially filled with aluminium cushioning, which can be removed when access to the reactor top is required.

The entire reactor is housed in a pressure-tight building having an inside diameter of 72 ft, and a total height of 120 ft, of which 51 ft is below ground level. The building, which has an ellipsoidal bottom and a hemispherical head, is built as a pressure vessel, of ASTM A-201 firebox quality Grade B steel, 1·03 in thick, double-butt welded, and fully radiographed. It has been pneumatically tested at 1·25 times the design pressure of 32 psi, and leak-tested to show leakage rates of less than 500 ft^3 in 24 hours.

Design conditions for the building (32 psi positive, 2 psi minimum, and wall temperature of 650°F) were assessed from theoretical studies and experimental work on possible accidents. It was assumed that a sodium leak could cause a fire completely consuming the oxygen in the building, and giving a maximum gas temperature of 1 240°F, with a wall temperature of 450°F, and all penetrations through the wall have gastight seals (located 18 in outside the wall so as to obtain the benefit of air cooling) designed for this service and each seal has been tested under simulated conditions.

A personnel air lock (8 ft nominal diameter) and an escape lock have been provided.

Shielding of the reactor can be considered in three stages; a primary shield that surrounds the core, secondary shielding that surrounds the primary shield tank, and biological shielding surrounding the complete plant.

First line of defence against neutron damage to the vessel is, of course, the neutron absorption in the blanket region. Outside this, the structure contains two rows of stainless steel bars, each the size and shape of a core element, giving a total thickness of about 5 in. Between these and the vessel wall are concentric layers of thermal shielding, cooled by the sodium flow, the total thickness, including the vessel wall, being about 7$\frac{1}{2}$ in. The irradiation of the vessel wall is expected to be limited to some $2{\cdot}5 \times 10^{21}$ nvt, equivalent 0·1 MeV neutrons; recent information shows no serious damage to stain-

less steel at $2{\cdot}5 \times 10^{22}$ nvt, so that there is a safety factor of about 10.

Outside the vessel there is a 6-in layer of 5% borated graphite, followed by a 2-ft 6-in layer of plain graphite and an outer layer of 1% borated graphite. A 3-in layer of insulation is placed between the inner layer of borated graphite and the remainder.

The neutron leakage flux at the outside of this shield is limited to 2×10^8 n/cm²s; this level is low enough to prevent damage to concrete. The gamma radiation is not expected to exceed 3 000 r/h at full power.

Around the primary shield tank is the secondary shield wall, 30 in thick and—very roughly—triangular in plan, which prevents the pumps, IHXs and secondary sodium loops from

Fig. 20.—General view of main control room.

direct neutron activation, reducing the neutron flux to about 10^4 n/cm²s. Both faces are clad with ¼ in steel and, opposite the primary loops, a 2½ in steel plate is added on the outside of the wall, with a 5 in air gap for cooling. This is not considered necessary on the inner face of the concrete, nor around the 30 in reactor outlets, which are provided with their own high-temperature shielding, within the wall, in the form of a self-supporting arrangement varying in thickness from 9-20 in, with anti-streaming collars. Calcium borate has been used, with a boron content of 11%; this material has a density of 70 lb/ft³ and a compressive strength of 800 psi after irradiation.

The biological shielding consists of the 5 ft concrete-and-steel operating floor, plus a 7-ft belt of concrete surrounding the containment building.

The reactor space beneath the operating floor is not normally accessible, and is sealed off below floor level. In order to prevent sodium burning, in case of a leak, the atmosphere will consist of dry nitrogen (max. O_2 content 5%) continuously circulated by external fans and cooled by heat exchangers. The total heat removed is $1{\cdot}26 \times 10^6$ Btu/h, and nitrogen is circulated at a rate of 30 000 ft³/min, the outlet and inlet temperatures being 130°F and 80°F, respectively.

Above the floor, a normal air atmosphere is provided. Allowable inside temperatures are from 40°F to 100°F and the external temperatures covered by the design range from −10°F to 100°F. Both heating and cooling is provided for the ventilation air which is recirculated, with a small direct intake of dehumidified fresh air. The building is held at a slight positive pressure.

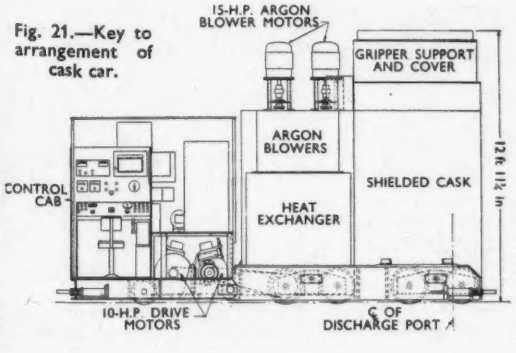


Fig. 21.—Key to arrangement of cask car.

Fig. 22.—View of cask transfer car from control end.

FUEL HANDLING

At each shut-down for refuelling, 11 new fuel elements will be placed in the reactor core and 11 irradiated elements placed in the transfer rotor to decay until the next shut-down. This enables transfer operations to commence immediately reactor conditions permit—it would obviously be very difficult to transfer irradiated elements directly from the core, with no decay period.

Transfer between the transfer rotor in the reactor and the fuel and repair building is by means of a self-propelled cask car (Fig. 21). This contains a storage rotor with 12 positions (11 elements and the exit port shield plug) and the necessary gripping and hoisting mechanism. The rotor is cooled by argon, circulated and cooled by blowers and an argon-air heat exchanger having a nominal cooling capacity of 100 kW. Elaborate arrangements are provided for the accurate location of the cask car over the reactor exit port, and for sealing the snout to the port, purging, and plug withdrawal.

The cask car arrives at the reactor with a full load of fresh fuel (previously warmed in an argon stream and contained in a sodium-filled finned pot) and exchanges elements in the reactor transfer rotor on a one-for-one basis. It then reseals the reactor exit port, and transports its load along a tunnel to the fuel and repair building. Both cask and heat exchanger compartments on the car are heavily shielded, as sodium vapour may cause Na^{24} to be deposited in the cooling circuit.

At the fuel building, the car discharges its load of irradiated elements, still in their finned pots, into a sodium-filled transfer tank which contains a 22-position rotor. From this, it is transferred to a cleaning machine. Here it is, for the first time, removed from its finned pot, being still argon-cooled. In the next stage, dry steam is mixed with the argon, in the correct proportion to react with the remaining sodium; after 1 minute, the argon is turned off and the steam continues for 4 min, followed by hot and cold water rinses. The element is then deposited in an underwater transfer car, and transported to the cut-up pool, where it is placed in a "leaker" container; after a short decay period the water is analysed for fission products. If activity exceeds 5×10^{-3} μc/cm³ the entire decay period is spent in the container, if not, it is removed; in either case it is transferred to racks in the adjacent decay pool. After 180 days decay, it is then returned to the cut-up pool, where blanket assemblies are loaded into shipping containers for transmission to reprocessing plants. Core elements are cut up to separate the core section from the axial blanket sections, as they will be processed at different plants.

Acknowledgment

Thanks are due to the Power Reactor Development Company for making available the latest (July, 1961) Hazards Summary Report on the Enrico Fermi plant, upon which this article is based.

Changing the Core Tube Nest of DFR under Operational Maintenance Conditions

By O. HAYDEN and J. PEARCE
(U.K.A.E.A Reactor Group)

THE first core tube nest of the Dounreay fast reactor* was designed to support annular fuel elements containing uranium metal of approximately 0·75 in o.d. After handover of the plant to the operating staff in December, 1958, and following a design study of larger fast reactors for commercial use it became evident that to achieve the high burn-ups required to produce economic power, cermet or ceramic fuels would be developed. The optimum design for these fuels is a thin pin or plate fabricated into sub-assemblies of up to about 4 in in diameter. In order to irradiate prototype sub-assemblies in DFR the original core tube nest was redesigned with a central hexagonal hole of about 5 in across the flats. Initially this hexagon will be charged with the tubular metal fuel elements, but at a later date they may be removed and replaced by three sub-assemblies of rhombic cross-section. On completion of the preliminary design work in July, 1959, the decision was taken to go ahead with the replacement of the original core tube nest. The reactor was due to go critical in November, 1959, and once this decision was taken it was important to make the change with the least delay; the longer the old tube nest was used the more radioactive it would become and the problems of shielding and remote handling would become more difficult. It was decided that the whole job should be done in six months, i.e. the handling equipment schemed, designed, manufactured and tested, personnel trained and the new core tube nest manufactured.

At this time the vessel and primary circuits were fully charged with NaK and this had to be kept circulating at low flow rates to prevent oxide build-up in stagnant sections of the circuits. No air could be allowed to enter the vessel and all operations had to be carried out through gas-tight equipment purged with nitrogen. The clearances between the tube nests and supporting skirt were small, and removal and replacement had to be carried out without disturbing this skirt; moreover, careless handling could cause jamming. Allowance had to be made for expansion and alignment of the handling equipment when heated and a grab had to be designed which would, without fail, pick up and hold the top tube plate of the nest. As the scheme evolved it was clear that special-purpose equipment was required which would positively and accurately control the position of the tube nest relative to the core skirt, and that careful planning of every detail would be necessary to avoid mishaps.

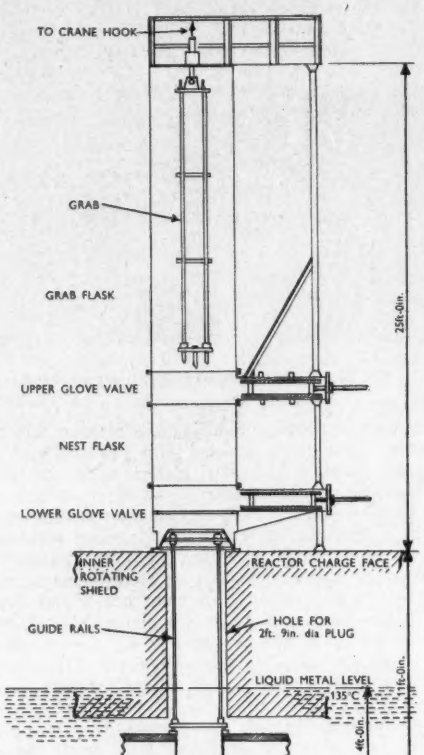


Fig. 1.—Diagrammatic layout of equipment during nest removal sequence.

Equipment

The equipment consisted of the seven main components (Fig. 1).

Lower Glove Valve. This was bolted on to the reactor top as shown in the diagram. It contained two large diaphragms which could be moved over the reactor top by using the rubber gloves attached to the walls of the valve. The lower diaphragm could be bolted into position on the reactor top flange so that the reactor was sealed. The other diaphragm could be bolted on to the base of the flask resting on top of the valve and so seal off the flask.

Plug Flask. This flask, which is not shown in Fig. 1, was made to fit on to the lower glove valve and to have sufficient height to contain the 2 ft 9 in diameter plug. This shielded plug is normally placed in a corresponding hole in the inner rotating shield and is bolted down, but it was removed prior to the actual operation of removing the core tube nest. The flask had a 3 in diameter lifting bar, the bottom of which could be attached remotely to the top of the plug, while the top passed through a gland in the flask and could be connected to the hook of the Goliath crane. This enabled the plug to be lifted into the flask without air getting into the reactor or flask.

Grab. This took the form of a long cage with a bottom flange to which were fitted three split collets which could be opened out by taper wedges operated by long screwed rods passing to the top of the cage and which were accessible via the rubber gloves. In the lowered position the collets of the grab, located by the guide rails, entered three holes in the top 2½ in thick plate of the tube nest and were opened out on the underside of the plate, so that on lifting the weight of the plate and tube nest was taken by the collets. The top of the grab was again attached to a lifting bar passing through the top of the grab flask and connected to the crane hook.

Guide Rails. In order that the grab located correctly in the old core tube nest and, more important, to ensure that there

* Dounreay Fast Reactor Symposium. Journal of the B.N.E.C., Vol. 6, No. 3, July, 1961.

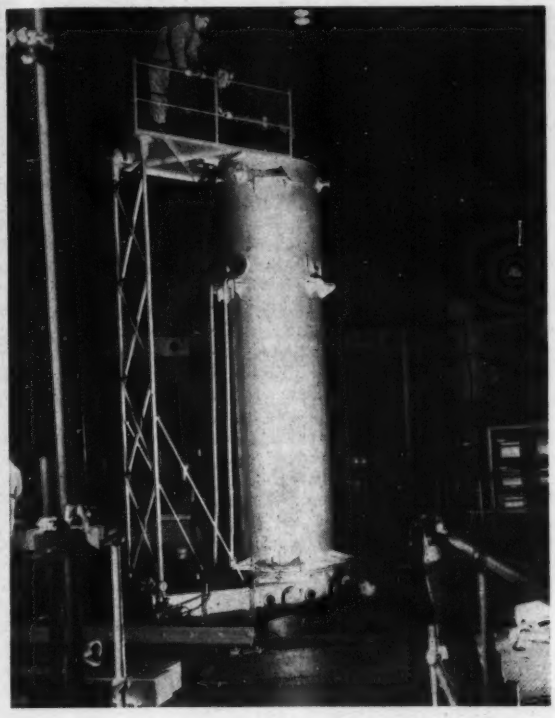


Fig. 2.—Equipment build-up on reactor charge face; lowering in the guide rails.

was no out-of-alignment during insertion of the new tube nest, three vertical guide rails were used. These were locked together top and bottom. The bottom end rested on top of the core skirt and was located in three breeder element holes, being accurately located by long tapered dowels. The tops of the guide rails were bolted to the top of the plug. This made the guide rail assembly rigid and prevented the core skirt from moving when the tube nest was lifted. The guide rails were inserted via the grab flask.

Grab Flask. This flask coupled-up to the lower glove valve was of sufficient length to take the grab or the guide rails and had a lifting bar going through a gland in the top of the flask.

Tube Nest Flasks. There were two of these flasks, one to contain the old core tube nest on removal and the other for the new core tube nest in the process of installation.

Upper Glove Valve. This valve was of similar design to the lower glove valve but positioned on top of either of the core flasks and underneath the grab flask. This enabled the grab to be sealed off in the grab flask and the core tube nest to be sealed off in the tube nest flask without air being allowed to get at these components contaminated with NaK.

Manufacture and Testing

It was arranged that the manufacturer of the new tube nest would collaborate in the design and subsequent manufacture and test the equipment required. An extensive full-scale test-rig set-up was necessary, involving lifts of six tons and a headroom of 60 ft, with simulation of the reactor top, 2 ft 9 in plug and tube nest. Routine pressure tests and leak tests were applied to all requisite items of equipment and tests carried out on the stuffing-box seal glands. Material test certificates and insurance company load tests were applied to all load-carrying items because of the serious consequences of any failure occurring whilst on the reactor.

Prior to the actual operation on site it was decided that two shifts would be used and a staff training programme was initiated. When the materials were received the test rig was erected in the Fuel Element Building adjacent to the sphere. The whole removal sequence was run through three times by each shift. The first time was to familiarize the men with the equipment and the way it worked mechanically, so no gloves were used and there was no concern about containment. The second time was to give the men experience working outside the flask with gloves and with restricted visibility through the flask and valve windows. The third time the bottom of the test rig was sheeted-in with sisal kraft paper so that conditions would be very similar to those on the reactor and the personnel could not see what was happening in the region of the core.

During the actual operation the reactor top was cleared and the lower glove valve was placed into position. A pressure test of this glove valve was carried out at 5 psi to ensure that a satisfactory joint was made on to the reactor top. This was important for once the plug was removed the valve formed the reactor seal.

The plug flask was put into position, connected up and pressure tested. The lifting bar was connected with a balance between it and the crane hook, and the plug was then raised at the crane's slowest speed. A man watched the balance for excessive load and gave directions to the crane driver. The plug came up into its flask and was removed from the reactor top to one side, being maintained in a nitrogen atmosphere inside the flask.

The guide rails were then put into the reactor, the tube nest flask and grab flask were mounted on the reactor, and the grab was lowered down on to the core. The grab was operated to grip the tube nest and the latter was then raised by means of a chain block, which was inserted between the crane hook and the lifting bar. This was introduced to give a sense of feel to the lift. The balance was watched all the time and the reading was approximately 1 ton for the whole lift.

The lower valve diaphragms were closed. The tube nest was then lowered until it rested on the bottom of its flask, and the grab was then released. The grab flask and the nest flask with the nest sealed inside it were removed from the reactor vessel. The grab was strippd, thoroughly decontaminated, then reassembled, new linear-type ball bushings being fitted.

At a later date the build-up was made on the reactor top for insertion of the new tube nest, and in general the procedure was similar. Care had to be taken to ensure that the nest was correctly orientated inside its flask, and no difficulty was experienced in lowering it and entering it into the core skirt. Finally, its position was checked by a form of dipstick, the grab released and withdrawn, and the grab and guide rails removed as one unit within the grab flask. Following this, the 2 ft 9 in plug was replaced and the operation completed. The complete operation took six days.

Although this was the first time that a heavy component fully immersed in NaK had been removed from the reactor vessel, the operation proceeded smoothly without a hitch. No great difficulty is foreseen in handling highly active components which will require additional shielding.

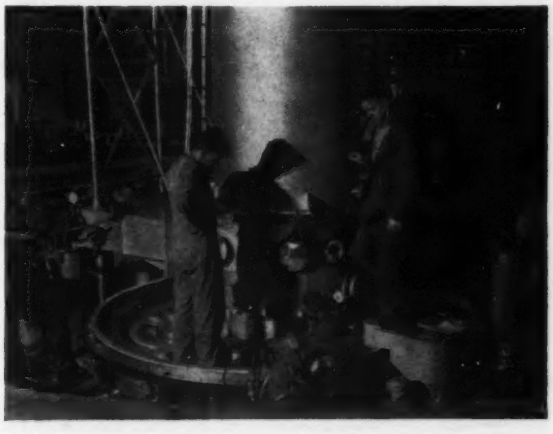


Fig. 3.—Lower glove valve on reactor charge face.

Powder Metallurgy

Report of the 1961 Plansee Seminar

A PLANSEE Seminar, entitled "Powder Metallurgy in the Nuclear Age" was held at Reutte in the Tyrol of Austria from June 20 to 24, 1961. This was the fourth Seminar held by the International Plansee Society, which was founded ten years ago by Dr. Schwartzkopf of the Metallwerk Plansee A.G., for the purpose of discussing recent advances in the field of powder metallurgy research and development.

The previous three Seminars have involved a total of 850 participants from over 20 countries who have presented some 200 technical papers. This fourth Seminar brought together 350 participants (and 100 wives) from 20 countries, ranging from the U.S.A., eastern and western Europe, India and Japan. At each Seminar, a new Chairman is elected to the International Plansee Society for a period of three years. The past Chairman, Dr. H. H. Hausner of New York, was succeeded this year by Dr. I. Jenkins, London. Also, at each Seminar, a plaque (of molybdenum) is awarded to the person who is considered to have contributed most to the understanding of powder metallurgy. This year, the honour fell to Professor J. T. Norton of M.I.T., U.S.A.

The opening address was delivered by Dr. Balke, Minister for Atomic Energy, Bonn, who stressed that personal contact was essential for successful technical progress. Dr. Balke then described how nuclear reactor exports were essential for the economic development of Germany, and how this in turn necessitated substantial effort devoted to nuclear research and development.

In all, 40 papers were presented in four days. As distinct from previous practice, the papers were issued as preprints, so that presentation at the meetings was restricted to 10 minutes per paper, thus conserving the maximum possible time for discussion purposes. Throughout the four days, the accent was placed on cermets and ceramics rather than on metals and alloys, due to the fact that greater attention was paid to nuclear fuels than to canning and constructional materials. Among the ceramic and cermet fuels, greater prominence was given to uranium compounds than to plutonium compounds. Apart from the final session on various aspects of refractory metals, the only canning material discussed was beryllium.

In the following paragraphs, a summary is presented of the various papers given and discussions held in the seven sessions which formed the Seminar.

Fuels (Metals)

Dr. H. H. Hausner, N.Y., opened the Seminar with a general paper on the requirements for metals in nuclear engineering applications. This paper formed an excellent introduction and contained extensive tables on the properties of nuclear engineering materials. Dr. Hausner stressed the point that in his opinion, materials and metallurgy constitute the bottleneck in present reactor development. In this connection, he emphasized that the achievement of adequate bonding between fuel and can is still one of the major outstanding problems which requires to be solved.

Baulbin and Jabot, CEA, France, described a pilot plant for the production of 1 kg/d of uranium metal powder. The process involved the conventional calcium reduction of UO_2 at a temperature in excess of the melting point of uranium. Excess calcium and calcium oxide are removed by acetic acid leaving a 95% yield of pyrophoric uranium powder of 15-44μm size (20μm average), containing as major impurities 0·5-1·0% oxide, 0·5 nitride and 1·0-3·0 hydride. In reply to questions, Baulbin stated that the uranium powder contained 100-700 ppm nickel and that the particles were spherical as distinct from the dendrites normally obtained by vapour phase deposition.

Jangg, Vienna Technical High School, described an amalgam method which had been used for the preparation of both uranium and zirconium powders. These amalgams are liquid at room temperature, and the mercury is removed by subsequent distillation. Powder of the desired particle size is produced by the suitable control of process variables.

Waldron *et al*, U.K.AEA, Harwell, presented a two-part paper on plutonium powder metallurgy. The first part dealt with the small scale preparation and consolidation of plutonium metal powder, from the hydride in high purity argon (< 20 ppm). Consolidation by hot pressing was superior to that by cold press-sintering, especially when the load was maintained whilst cooling from the delta phase. The density thus obtained was 19·1 g/cm^3 (19·5 g/cm^3 for cast metal). The delta to gamma transformation results in disruption of the inter-particle bond.

The second part of the Harwell paper dealt with early work on the development of PuO_2-UO_2/stainless steel cermets of 50% volume mixtures as a fast reactor fuel. The best dispersions were obtained using ceramic particles of 200-300 μm size. Cold pressing and sintering gave only 75% density compacts. Triple pressing improved this value to 80%. The best results were obtained by hot pressing at 800-950°C, which gave values as high as 95% density. Further work on hot pressing is still in progress. In addition, swaging, rolling and extrusion are under development for improved fuel-to-can bonding. In reply to questions, Dr. Murray stated that the poor density from the cold press-sintering of plutonium powder was not due to surface contamination by graphite. The interrogator (from France) emphasized however, that a reaction did occur between graphite and uranium powder at 450°C, and that the reactant film was not detectable by X-ray or metallography.

Fuels (Oxides)

A paper by Kuczynski, University of Notre Dame, U.S.A., dealt with the effect of oxygen on the sintering of oxides. Oxides sinter faster than metals at equivalent temperatures relative to their melting points. Work on Al_2O_3, Fe_2O_3 and ZrO has shown that ambient oxygen may increase significantly the apparent diffusion coefficient in oxides, and thus increase the sintering rates. This is attributed to the increase of vacancies due to the oxygen adsorption. The authors indicated that the stresses present at the neck between two particles are able to reduce UO_{2+x} to almost pure UO_2, and so change the diffusion mechanism causing a reduction in sintering rates.

During the discussion, Dr. Rhines (University of Florida) stated that the diffusion coefficient was very sensitive to particle size. The surface tension is very high for small particles in the neck area. Work with wire samples showed that the rate is higher for smaller diameter wires. In reply, Kuczynski stated that his original work with copper wires did not confirm Dr. Rhines' data. Dr. Harrison (Harwell) said that his work on the sintering of PuO_2 pellets confirmed the authors in that sintering proceeds by a diffusion mechanism as in Al_2O_3. An M.I.T. representative stated that a comparative study of single and poly crystals indicated that enhanced diffusion occurred at grain boundaries. He concluded that our understanding of the effects of surfaces in powder metallurgy is very limited.

Gebhardt and Elssner (Max Planck Institute) spoke of the reaction between UO_3 and ZrO_2. The results on the study of this system vary widely, and the present work was intended to elucidate the position. Mixed powders of varying grain size were compacted and annealed at 1 650°C before X-ray parameter measurements at room temperature. The effects of particle size and annealing temperature on attaining equilibrium were determined by parameter measurement. Further work is needed to complete the system. Someone (from Germany) stated that in other systems they had found that the rate of quenching from the annealing temperature was important.

Harrison *et al* (U.K.AEA, Harwell) presented a very interesting paper on a study of the sintering of mixed PuO_2 and UO_2 up to 25% PuO_2. Density, oxygen-to-metal ratio and degree of solid solution attainment were correlated with Pu/U ratio,

and other variables such as sintering atmospheres, temperature and time, compaction and powder preparation methods. During the discussion, a Belgian stated that he had followed the course of sintering at 1 000°C (in Ar and H_2) by microphotography at temperature. The results showed that the addition of PuO_2 up to 20% reduced the sintering rate of UO_2. In reply to a question, Harrison gave his particle size as—UO_2, 5 m²/g; PuO_2, 20-30m²/g. Some results were unusual in that the density decreased with time at temperature and with increase in sintering temperature. The effects of complete densification or attainment of solid solution on irradiation behaviour have not yet been determined.

One of the main papers was that of Cunningham *et al* (Battelle Memorial Institute) on the preparation and properties of cermet fuel elements containing 60-90% UO_2 in matrices of stainless steel, Nb, Mo and Cr. Work is now in progress at Battelle on cermets of 60-90% carbide and nitride, as well as on cermets containing 50% UO_2, i.e. dispersion fuels. Fabrication included hot isostatic pressing, while both physical (electrical and thermal conductivity and linear expansion) and mechanical (rupture and compressive strengths and elastic modulus) properties were evaluated. The speaker brought out several interesting points: plastic flow does take place before fracture with these cermets. Nb coated UO_2 particles (obtained by vapour deposition) gave a good can bond, which promotes the best resistance to thermal shock. The authors readily obtained uniform dispersion with 80% UO_2 in stainless steel. In conclusion, the authors stated that no fuel could be truly assessed without determining irradiation behaviour. This is now in progress.

In reply to questions, Cunningham stated that hot pressing was carried out with He at 10 000 psi. The effect of additions to improve the thermal conductivity of UO_2 seems unprofitable because the conductivity is ionic and the electronic contribution is negligible.

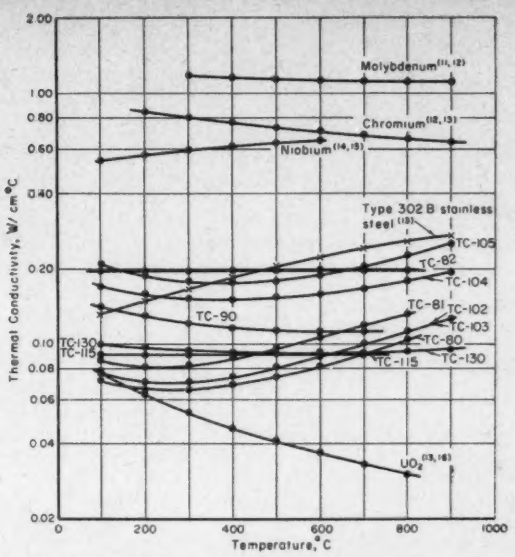
Dai *et al* of Japan presented a paper on ceramic and swageable UO_2. This dealt with the effects of process variables on UO_2 powder properties produced by A.D.U. method, and also with the effects of high temperature reduction of UO_2 on UO_2 powder prepared for consolidation by swaging. The results showed that powder properties are dependent on reduction temperature as well as on calcination temperature. In addition, oxidation reaction is dependent on the nature as well as on the extent of the surface area. Particle size is determined by the reduction temperature and the parent particle size. Sintering behaviour (i.e. final density and shrinkage) can be closely controlled by the correct choice of powder preparation variables. The average particle size of the powder was 0·1 m²/g; melting after pre-sintering was not used as it was considered an inefficient method of powder production; several tons of UO_2 powder of good quality have been produced; present efforts are devoted to improved control of sintering shrinkage, so as to avoid grinding losses.

Dr. P. Murray of Harwell briefly described work on the preparation of single crystals of UO_2 by the electrolysis of uranyl chloride in a fused alkali chloride using a platinum cathode and carbon anode. The crystals of 3 mm diameter exhibited a density of 10·85 g/cm³ with an oxygen/uranium ratio of 2·02 and 300 ppm total impurity. The crystals are being used for deformation study and the determination of fission gas release at high temperatures.

Morgan of Oak Ridge had studied the effect of powder variables on the sintering of thorium oxide. He had found that sinterability decreased with increase in powder size, with increase in powder density and with increase in particle size after varying blending treatment.

Fuels (Oxides, Cermets)

Four papers were presented in this session, of which the most interesting was that by Delmas and Hauser (France) on new developments in the technology of UO_2. This paper was in two parts. The first described the preparation of UO_2 powders in batch and continuous laboratory furnaces. UO_3 (obtained by the calcination of ammonia uranate at 380°C) was reduced with cracked ammonia at 680°C to give a powder of 4 m²/g specific surface. Powders of higher reactivity with a specific surface of 10 m²/g were also prepared.



Cunningham *et al*, thermal conductivity of cermets against temperature. TC 80, 102, 103 : UO_2/-20 v/o s.s.; TC 81 : UO_2/-30 v/o s.s.; TC 90, 105 : UO_2/-20 v/o Mo; TC 82 : UO_2/-30 v/o Mo; TC 104 : UO_2/-20 v/o Cr; TC 115, 130 : UO_2/-20 v/o Nb.

The second part of the paper dealt with the fabrication of bars and tubes (15 mm dia. × 260 mm long) in UO_2 by cold extrusion followed by sintering in cracked ammonia at 1 600-1 650°C. Densities of 10·25 g/cm³ were obtained, which compared with densities up to 10·3 g/cm³ obtained by isostatic pressing. Extrusion proved cheaper than normal compaction methods. The work also involved the sintering of mixtures of UO_2 and ThO_2 in air to densities in excess of 97% theoretical. In addition, oxides of certain metals were added in small quantities to UO_2 in order to increase the thermal conductivity. Homogeneity is an important requirement, and this was best obtained by multiple-hole extrusion of the powder mixed with a binder. Extrusion pressures were of the order 1 ton/in² at a speed of 1 m/s.

Dr. Halle of Sweden described his work on the sintering of 15 tons of UO_2 powder to 70 mm diameter pellets. Urea precipitated powder was mixed with a lubricant, cold pressed at 5 ton/in² and sintered at 1 700°C for five hours. This gave a density of 10·6 g/cm³ with <95% efficiency. Results indicated an increase in specific surface with decrease in reduction temperature. The final density increased with increase in compaction pressure.

Dr. Runfors of Sweden spoke of grain growth in UO_2 pellets and stated that grain growth was undesirable because of fission gas release during grain boundary movement. Isothermal grain growth at 1 500-2 200°C in moist hydrogen does not proceed rapidly until the porosity is less than 4%. Grain growth during thermal cycling from 1 450° to 1 550°C increases with the number of cycles. A sectional temperature gradient from the melting point of UO_2 at the centre to 1 800°C at the surface caused a grain size of 1 mm at the centre with the formation of a white phase (uranium metal).

Dr. P. Murray (Harwell) described recent data on the release of fission gas from UO_2 based on a survey of 60 fuel elements. The survey was restricted to elements >90% density with oxygen/metal ratio <2·04. The results showed that fission gas release increased with increase in centre temperature of the fuel bar from 900° to 2 000°C. Low density and departure from stoichiometry did not appear to affect the gas release, which is difficult to understand. Dr. Murray stressed the fact that it is necessary to obtain an understanding of fission gas release at temperatures in excess of 1 600°C in order to take full advantage of UO_2 fuels.

Gurland of Brown University, U.S.A., gave an interesting account of a quantitative evaluaton of the structure of a 2-phase system based on silver-bakelite mixtures. Statistical theory of the spacing was supplemented by experimental data using electrical conductivity measurements.

Lenel, Rensselaer Polytechnic, provided a very useful literature review of dispersed phases in the powder metallurgy of nuclear materials. The review covered two types of dispersed phase systems. The first dispersion type fuels consist of a

fissile phase (100-400 μm) dispersed in a non-fissile matrix. The second, covers the dispersion of fine particles ($<1\ \mu$m) in the matrix for strengthening purposes. Specific applications were discussed.

Caillot (CEA) mentioned SAP type magnesium developed for fuel cans by oxidizing magnesium powder in CO_2 at 400°-450°C. This material may find use in the EL4 reactor.

Uranium Compounds

Ten papers were presented during this session, some of which concerned plutonium compounds.

Malterson *et al* (Plessey Co.) described the preparation and property evaluation of the tetraborides of Th and U and the diboride of U. A short review of the silicides and carbides of uranium was included. Adamson *et al* (Oak Ridge National Laboratory) presented an extensive paper on the fabrication and radiation performance of dispersion fuel elements of the type U_3O_8 + Al; UO_2 + Al; UC_2 + Al as well as U-Al alloys. Burn-ups as high as 80% were achieved. Oxide dispersions showed little volume change, whereas the UC_2 + Al fuel exhibited severe swelling.

Keller and co-authors of Battelle, described their work on the preparation and consolidation by cold-press-sinter and hot isostatic pressing of UC and UN powders. During vacuum sintering, free uranium was formed in all cases, while acceptable densities were obtained only by sintering in the presence of a liquid phase. Densities >95% were achieved by hot isostatic pressing at 5 ton/in^2 at 1 450°C.

Hedges and Regan (Harwell) spoke about the properties of UC prepared by the direct reaction of the component powders, i.e. uranium and graphite. These properties included thermal and electrical conductivity, hardness and rupture strength. The ability for fission gas retention was relatively high.

Four other papers dealt with uranium carbide. Borta *et al* (Sheffield University) described their work on the phase diagrams of UC with certain transition metals, viz., Cr, Fe, Ni and UFe_2. A second paper by Brown (C. A. Parsons Co.) was concerned with the preparation and properties of UC, while a third by Caillot (CEA) described industrial methods of producing dense UC, namely direct current heating and electron beam melting. The fourth paper, by Himmelstein (Germany), described the preparation of UC by vacuum sintering and by arc melting. Two papers only were concerned with plutonium compounds. One from Los Alamos Scientific Laboratory was entitled: The Powder Metallurgy of Plutonium Fuel Materials, while the other, by Pascard (CEA) described work on oxide and carbide solid solutions of uranium and plutonium.

Construction Materials

In this session, three papers dealt with beryllium while the remaining two were on metal borides and pre-alloyed powders.

The first of the beryllium papers (by Syre of Pechiney Co.) described the application of powder metallurgy techniques in the processing of beryllium products (ingots, tubes and discs). Two preferred processes, sintering-forging and sintering-drawing, gave good properties combined with high densities. The second paper, by Martin and Ellis (AWRE Aldermaston) was a critical survey of the techniques used for the consolidation of beryllium powders. Of particular interest was the mention of current development at AWRE of formable sheet by the hot sheath rolling of beryllium ingot metal. The authors advocated more attention to ingot beryllium for the attainment of better ductility and lower costs. The third paper was by Rich and Walters (Harwell) on the effects of irradiation on beryllium at temperatures up to 350°C and to a maximum fission dose of 2×10^{20} nvt.

Holden *et al* (GEC, U.S.A.) spoke of their work on the preparation and properties of the diborides, tetraborides and hexaborides of the rare earth elements. Radiation swelling is the main problem in these brittle materials; its mechanism is difficult to understand. Sands and Oliver (B.S.A. Group) discussed the preparation, compaction and properties of pre-alloyed powders. The properties, with the exception of elevated temperature fatigue, were all superior to those of the fabricated ingot alloy. Fitzer (Siemens Plania) described the preparation of non-porous graphite, by liquid and gas impregnation, for reactor cladding applications. Work on blocks, tubes and sheet has shown impermeable graphite to be available. Burn up resistance has yet to be solved.

Technology

The first paper by Sari and co-workers of Euratom covered the coating of UO_2 particles by Mo, Cr, Ni and Fe by vacuum metallizing to a depth of 2-15 μm. The coatings have not as yet been evaluated by thermal cycling or by irradiation. The object of the work was to prevent reaction between PuO_2 particles and the matrix, and to contain fission gas. Alloy coatings have also been prepared.

Arbiter and Stern (Nuclear Development Corporation) discussed the preparation and properties of oxide coated uranium powder compacts prepared by hot pressing with subsequent extrusion. A 20% oxide content gave the same hot hardness values as U-Mo alloys. Irradiation testing is now in progress at temperatures >500°C, the present limit of unalloyed uranium. Dr. Murray stated that similar work at Harwell showed the material to possess poor irradiation behaviour. Tool failure had prevented extrusion in the top alpha temperature range.

Garlé *et al* (CEN) described two methods for the production of dense spheroidized particles. The first, used for plutonium carbide, involved mixing with camphor dissolved in methyl alcohol followed by treatment on a vibrating sieve. This resulted in >95% particle density. The second, used for uranium carbide, involved plasma jet melting, but it tends to produce a central hole in the particles. Dr. Murray (Harwell) stated that the plasma method has been used successfully for particles of Mo, W and MgO.

Two other papers dealt with extrusion. The first by Meny *et al* (CEA) concerned the extrusion sintering process for the preparation of U-Al and UO_2—stainless dispersion type fuel elements. Bars and tubes 10-30 mm dia. × 1 mm. were produced to almost theoretical density. The second paper (Loewenstein of Nuclear Metals, Inc.) presented a survey of the hot extrusion of various nuclear materials, i.e. U, Th, Be.

Schneider (NUKEM) described the consolidation of thorium metal powders to high densities without the necessity of final sintering. This is possible because of the strong cold welding tendency of the metal. Aluminium foil prevents powder-die tool seizure.

High Melting Metals

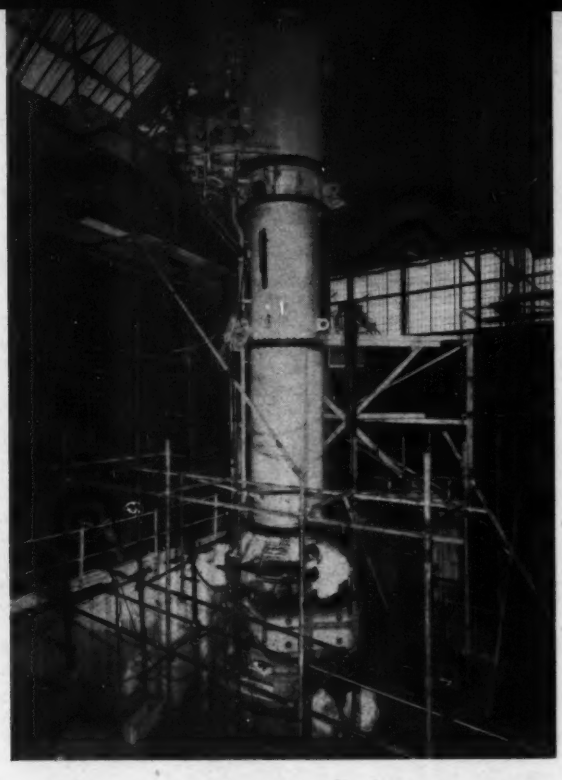
This was the final session in which six papers were presented, dealing mainly with metals and alloys as distinct from ceramics.

Deutsch and Ault, N.A.S.A., Washington, spoke on powder metallurgy materials for rockets, missiles and other space vehicles. These materials included light metals, heavy metals and cemented carbides. Much emphasis was placed on beryllium, the most widely used metal at present for space vehicles. Other materials included composites of fine powders and fibre dispersions.

A most useful paper was that of Jaffe *et al* (Battelle) on the effects of dispersions on the ductile-brittle transition and recrystalline behaviour of tungsten. This concerned sintered W-base alloys with up to 10 v/o ThO_2, ZrO_2,HfO_2,Al_2O_3, UO_2, SiO_2, MgO, TiN, HfN, TaN, HfC, TaC, Ta_2C. These could be fabricated to strip. Primary recrystallization was raised by 200°C by certain compounds, while others retarded secondary recrystallization to 2 700°C. The ductile-brittle transition was lowered as much as 90°C.

Brooks and Harris (Cavendish Laboratory) described their work on the tensile properties of Nb, Ta, W, Mo and Ir up to 2 200°C. The mechanism of deformation and fracture was found to be similar to that of the lower melting point metals at homalogous temperature. The solid solution alloys of Pu with Mo were described by Geach and co-workers of AEI, Ltd. They described the fabrication, hot hardness and corrosion resistance properties. Goeltzel and Landler (New York University) discussed silicide coatings for the surface protection of tungsten up to 1 800°C. Intermediate layers of ZrO_2 were investigated, but showed no advantage. A brief but most interesting paper was presented by Mitsche *et al* (Leoben) on the application of the Reichert microscope to the study of the sintering of various materials at temperatures up to 2 000°C. Photographic recordings were obtained. This technique should prove most useful in the study of sintering mechanisms.

Industrial Notes



Control rod and actuator machine for fuel handling operations at Berkeley (see *Nuclear Engineering*, April, 1960) being assembled at the Wolverhampton works of the John Thompson Ordnance Company for pressure testing before being transferred to the station in sections.

Two power manipulators, each with a lifting capacity of 750 lb, and a 1½-ton hoist are to be supplied by the **GEC** to the AEA for installation in the decanning and decontamination caves at Windscale. The three units will share a common set of rails, the hoist being placed between the manipulators, and it will be possible to operate them either separately in different caves or together in one.

During July the **Plessey Company's** new mobile showroom toured nuclear power station sites and research establishments. Comprising an articulated 10-ton tractor-"caravan" trailer unit it housed a display of the company's latest developments in the nuclear instrumentation field.

The functions of the four specialized sections of the **Institution of Electrical Engineers** covering electronics and communications, measurement and control, supply and utilization, have been taken over by three new divisions. Two of these represent electronic and power interests and the third—a general division—covers activities common to all electrical engineers, such as basic measurement and technological education.

The licence agreement between **Brown, Boveri and Company** and **Richardsons Westgarth** covering the manufacture and sale in the U.K. of Brown Boveri turbine equipment has been terminated. Such equipment will now be manufactured by the Swiss company and marketed through their U.K. company, British Brown-Boveri, 75 Victoria Street, London, S.W.1. The two companies will continue to collaborate for the completion of current contracts as well as the servicing of existing plant, including the supply of turbo alternators for Trawsfynydd.

John Thompson have opened a new office building in Woburn Place, near Tavistock Square, London, to house the sales offices of the group's 17 home companies. Previously they were situated at Imperial House, Kingsway, and in other parts of London. The new building forms a new east wing to the extensive Tavistock House of the British Medical Association.

Tannoy have installed four communication and warning systems at Windscale for use in connection with operation of the AGR.

Johnson, Matthey and Company announce that they are now able to supply scandium, yttrium, and most of the other 14 rare earth metals in strip form in thicknesses down to 0·001 in and with a maximum width of 3 in. This is the latest development in the company's production of these metals which have previously been available as ingots and as wire of diameters down to 0·02 in.

C. A. Parsons and Co. have signed a licence agreement with **Zallea Brothers,** Delaware, U.S.A., for the manufacture and sale of thin walled bellows expansion joints of Zallea design. The agreement provides for sales in the U.K., EFTA and British Commonwealth, except Canada.

Applications of radioisotopes were featured in the **Radiochemical Centre's** exhibit at the fifth international congress on biochemistry, Moscow, August 10-16.

Hilger and Watts, manufacturers of scientific instruments, have formed an association with **Wray (Optical Works).**

A new company has been formed by the vibration division of **W. Bryan Savage,** a member of the Pye Group, and **Ling Temco Electronics,** of Dallas, Texas. Called **Pye-Ling,** it will make available a comprehensive range of vibration testing equipment.

A chemical and nuclear engineering division has been set by **Uddeholms Aktiebolag,** Sweden, for the construction of heavy steel equipment such as pressure vessels and fractionating columns. Design work and orders for the new division will be carried out at the company's works at Degefors—Degefors Jarnverk—where one of the current larger pieces of equipment being built is the reactor tank for R3/Adam. United Kingdom sales and service for the division are handled by the London office at 78 Buckingham Gate, S.W.1.

The Atomic Industrial Forum and the **American Nuclear Society** have announced that their joint nuclear conference and trade fairs for 1962 and 1963 will be held in Washington and New York respectively. This year's venue is Chicago during the week beginning November 5.

A second top closure for the *Savannah* reactor vessel is to be made to accommodate new control equipment now under development by the **Borg-Warner Corporation.** It will be designed and fabricated by **Babcock and Wilcox,** New York, main contractors for the design and construction of the reactor system.

General Electric, San Jose, California, are to supply burst slug detection equipment for the EGCR at Oak Ridge and a neutron monitoring system for the new production reactor at Hanford. Total value of the contracts is $200 000.

Tracerlab are to merge their interests with those of **Laboratory for Electronics,** one of the top 100 companies in the U.S. Combined annual sales of the two companies are valued at $60 million, combined payroll amounts to 4 000 and together the two companies have sales offices and representatives and manufacturing subsidiaries in 32 countries.

The U.S. AEC has signed a contract with **General Electric** for the use of the company's engineering test reactor at Pleasanton, California, for test irradiation and related services. The contract is effective for one year and may be extended for a further year at the option of the AEC. Payment for the service will be on a unit price basis. Hot cell and other experimental support work related to the irradiation service will be paid on a cost-plus-fixed-fee basis.

Allis-Chalmers Manufacturing Company have been awarded a $341 570 contract for the construction of a 23½ ft high, 8 ft dia. stainless-steel clad reactor vessel for the High Flux Isotope Reactor (HFIR) to be built at Oak Ridge. ("World Digest," June, 1961.)

New Films: "Atomkraftwerk Kahl," about Germany's first nuclear power station with commentaries in German, English, Spanish and Italian. Available from AEG, Frankfurt/Main. "Matter in Question," about CERN. In colour and runs for 25 minutes. Available in either 16 mm or 35 mm for non-commercial screening from national information centres of member countries or from Public Information Office, CERN, Geneva. "Technical Information Services of the AEC." In colour and runs for 20 minutes. Available from U.S.AEC's home and overseas film libraries.

The Lead Development Association have moved to new offices at 34 Berkeley Square, London, W.1. Telephone number is Grosvenor 8422.

C. A. Parsons and Co. have moved their London office to a new suite of offices at 40 Broadway, Westminster, S.W.1.

Dragon Project Report

The design of Dragon was reconsidered during the second year of the project's implementation and requirements of specific items assessed. It was found unnecessary, however, to make any fundamental changes in the overall design (*Nuclear Engineering*, July, 1960) according to the recently published annual report covering the period ended March 31, 1961, but improvements have been made in certain specific items as, for example, the design of the fuel element spikes, the fuel boxes, the main shield plug and in the fission product plant.

The manufacture of the pressure vessel proceeded satisfactorily and experimental results indicated that a high standard of leak tightness will be obtained. The original idea of having a Dragon-designed prototype circulator was abandoned because of the limited time available. However, an extensive series of tests in support of a circulator design submitted by a contractor was started. Manufacture of the charge machine began during the period under review and, apart from certain simplifications introduced to improve the reliability of the machine, no fundamental changes in design were made.

The containment and ventilation of the reactor building were completely finalized and they follow closely the scheme described in the first annual report. Also with relatively little change are the main safety aspects attached to the project.

During 1960-61 contracts with industry to the total value of £4 million were negotiated. Expenditure is estimated at £2 379 000 and the budget for 1961-62 is £4 703 000. No change is foreseen at present in the total estimated expenditure of £13 680 000 during the life of the project.

The build up of staff continued and nearly reached full strength, put at 265.

On the research and development side heat transfer studies continued on the three rigs at Winfrith and a contract was placed for the construction and operation of a single reactor channel high-temperature heat transfer rig. Construction of the seven fuel element rig was completed and put into commission. It was decided not to proceed with renting the mass transfer rig being built by the AEA, but instead to carry out relevant experiments on smaller rigs.

Purification Process Changes

Major changes have been made in the fission product and helium purification process, which was in the first place only of a provisional nature. One of these changes introduces the oxidation of impurity gases such as Co, H_2 and CH_4 to H_2O and CO_2 which are easier to trap, the trapping being carried out by freezing the H_2O and CO_2 on cold surfaces in a freezer unit as originally planned. Another change is that no attempt is made to separate the stable and long-lived Kr and Xe from the non-oxidizable impurities for final disposal. These changes arose from a re-evaluation of the data for adsorption of gases at liquid nitrogen temperature which indicated that it would not be feasible to separate the xenon and krypton from CO, H_2, CH_4, N_2 and A, and that these impurities would interfere seriously with the adsorption of xenon and krypton on charcoal. Accordingly it was considered expedient to oxidize and trap out CO, H_2 and (if possible) CH_4 before the low-temperature charcoal traps.

Changes in the fuel element include the shortening of the fuel element spike so as to provide space for the fission product traps in the bottom ends of the central rods and to provide a better bottom reflector to improve the reactor physics of the core. A change is envisaged in the live reflector, as recent evidence indicates that the long graphite rods required by the present design may bend under high irradiation and temperature conditions and possibly interfere with control rod movement. In the static reflector, cooling is now to be achieved by circulation of the helium through a series of circular holes on a common pitch circle diameter instead of through a central annular gap in the reflector.

Personal

Appointments—U.K.

Mr. J. N. V. Duncan, managing director of The Rio Tinto Company, as chairman of Nuclear Developments, the company recently formed by ICI (Metals Division), Rolls-Royce and Rio Tinto, for the preparation and manufacture of nuclear fuel. Other directors are **Mr. R. W. Wright,** deputy managing director, Rio Tinto; **Sir Mark Turner,** director, Rio Tinto; **Mr. J. D. Pearson,** chief executive and deputy chairman, Rolls-Royce; **Mr. A. A. Rubbra,** director, Rolls-Royce; **Mr. L. Barman,** special executive, Rolls-Royce; **Dr. J. Taylor,** director, ICI; **Mr. St. J. de H. Elstub,** chairman, ICI Metals Division; and **Dr. R. L. P. Berry,** director, ICI Metals Division. **Mr. S. S. Smith,** research manager, ICI Metals Division, becomes executive director and general manager of the new company, and **Mr. P. J. G. Elwes,** of Rio Tinto, secretary.

Sir John Cockcroft, as first president of the amalgamated Institute of Physics and The Physical Society.

Mr. F. W. Stokes, general manager of Powell Duffryn Carbon Products, as managing director.

Mr. H. C. H. Matthews, as technical director of Pantak.

Mr. C. J. Powers, of York Shipley's industrial department, as assistant manager of the contracts division of the department.

Mr. L. C. Dawson, head of the Cape Asbestos group engineering division, and Dr. R. Gaze, the group's chief scientist, to the board.

Mr. G. Bernard Feltbower, formerly general manager and chief engineer of the Electro Dynamic Construction Company's control gear division, as chief engineer of English Electric's control gear division.

Mr. N. Kurruish, consultant mathematician and lecturer in AEI's engineering mathematics department, as manager of the department. He succeeds **Dr. W. J. Gibbs,** who has relinquished the post but remains a group company consultant.

Mr. Henry F. Sherborne, managing director of Yorkshire Imperial Metals, as president of the British Non-Ferrous Metals Federation.

Mr. W. G. Harrison, general manager of Dynatron Radio, to the board. Mr. Harrison joined the company three years ago as sales manager.

Mr. J. F. D. Wood, general sales manager of Sanders and Forster, a member of the Chamberlain Group of civil engineering and property companies, to the board as sales director.

Appointments—Overseas

Mr. Lloyd B. Kramer, manager of space electrical power systems development work for Westinghouse, as manager in charge of nuclear reactor core developments.

Mr. Leonard J. Linde, director of Allis-Chalmers engineering services, as director of engineering of the company's industries group.

Dr. Clifford E. Weber, as director of fuels and materials for Atomics International. Before joining AI in January of this year Dr. Weber was for 15 years at General Electric's Knolls atomic power laboratory, Schenectady.

Obituary

Nuclear Engineering regrets to announce the death of:—

Dr. G. C. Dalton, director of the Australian AEC research establishment at Lucas Heights and formerly a principal scientific officer at Harwell. A victim of poliomyelitis while at Oxford in the thirties, Dr. Dalton was only 45 years old at his death.

Mr. T. C. Hale, joint managing director of the Cape Asbestos Company, after a long illness. He was 57.

Technical Papers and Publications

SHIPPINGPORT STORY

Selected Operating Experience of Commission Power Reactors, by J. O. Roberts (Division of Reactor Development, U.S.AEC). (Paper for the Summer General Meeting of the American Institute of Electrical Engineers.)

This paper, in which several reactors are reviewed, is directed more towards those interested in the operation of reactors rather than in their design. In summarizing, attention has been focused on the Shippingport plant as it operates more nearly as a commercial plant than any of the others. When considering the operating figures of Shippingport, however, it is essential to bear in mind its importance, as a pioneer, in experimental and training work; it cannot be regarded simply as a commercial plant.

Shippingport (*Nuclear Engineering*, June, 1958, p.233) went critical in December, 1957 and, during the next two years operation totalled 5 806 full power hours, generating some 388·5 million kWh. Replacement of the first seed of the first core began in November, 1959, and the reactor returned to full power on May 7, 1960, after five months of refuelling and one month of testing. By the end of 1960, the total output had reached 650 million kWh.

Since the plant has four primary loops and is capable of full-power operation on three, plant availability has been high, in spite of problems with pumps and steam generators. It has also demonstrated that the PWR concept is capable of operating on either base or peak loads; it is stated that in flexibility and response to load changes it is superior to fossil-fuelled plants. Peak load operation and the ability to take load swings falls off with core age, of course, as the excess reactivity available to override transient xenon poisoning is reduced.

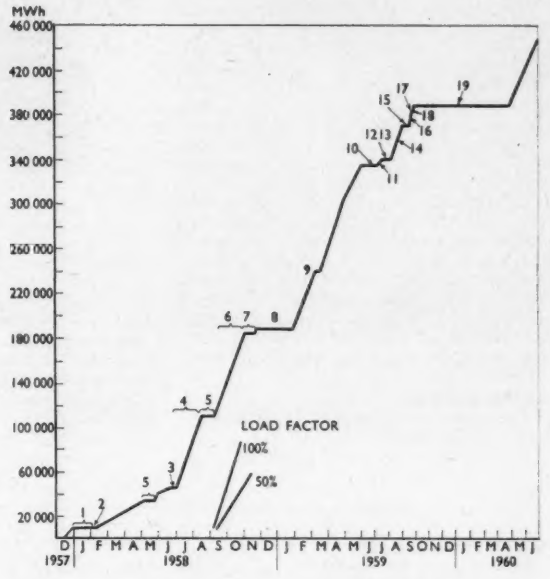


Fig. 1. Operation chart for Shippingport. Cumulative Gross Electrical MWh from start-up to July 1960.

KEY

1. Reactor shut down for test and training.
2. Leak discovered in steam generator.
3. Coolant loop pump failed.
4. First 1 000-h reactivity lifetime test.
5. Shut down for maintenance and testing.
6. Second 1 000-h reactivity lifetime test.
7. Shut down for maintenance and testing.
8. Moisture separator failure discovered; plant remained shut down through January 1959.
9. Repair of turbine governor.
10. Periodic operation for training purposes.
11. Shut down for maintenance and testing.
12. Periodic operation for testing plant dynamic response.
13. Shut down for capacity reduction and hot subpower tests.
14. Final reactivity lifetime test. Final run at full power for Core 1, Seed 1.
15. Low-power physics tests.
16. 50-MW operation for end-of-core-life tests.
17. Shut down because of xenon build-up.
18. Final power run at 20 MW.
19. Shut down to replace core seed, full power resumed.

Shut-downs

Fig. 1 shows the general pattern of operation; it does not indicate all of the interruptions. An interruption is defined by the Duquesne Light Company as an occurrence which (a) results in loss of station output; (b) jeopardizes personnel or equipment safety; (c) interferes drastically with plant operation; (d) results in a safety shut-down.

During the two years of operation with Seed 1, there were some 90 interruptions, of which 23 resulted in safety shut-downs, 12 occurring during power operation, resulting in a loss of electrical output. Analysing these, it was found that 8 were caused by equipment and 15 by personnel error. Of these 15, equipment design contributed to one, test operating contributed to two, and maintenance the same. Improper operation of test equipment was responsible for five, while the remaining five had no contributing factor.

During the second year, safety shut-downs were about half those in the first year—attributable to better understanding of plant restrictions and familiarity with control circuitry. The ratio of personnel-to-equipment causes remained about the same.

Major Maintenance

Two major maintenance jobs, involving pump removal and inspection of heat exchangers for tube leaks were carried out; both were successfully completed but required working in relays due to dose accumulation, the level at the pump piping being about 25 mr/h and, inside the heat exchanger 2·5 r/h. The mechanical work was very small compared with the many man-hours spent in clean-up, shielding and precautionary measures. The pump, which had to be returned to the manufacturers for repair, had to be decontaminated to less than 100 dpm/ft² beta-gamma per smear sample; this took two months. The time for decontamination of components has now been greatly reduced by the installation of an ultrasonic bath.

Core Components

Performance of fuel elements has been good. Blanket assembly F-2, suspected of having one or more defects was removed and replaced; the reactor is designed to run with defective rods and this was carried out more as a demonstration of the efficiency of the FEDAL (Failed Element Detection and Location) gear than as an essential operation.

Two blanket elements were dismantled under water and a single bundle replaced in each, showing the flexibility of design. Sections cut from the blanket fuel showed no distortion and a maximum of 0·11% fission product gas within the cladding. The absence of grain growth indicated that centre-line temperatures were less than 2 650°F and that heat flux during the operation of Seed 1 was less than 300 000 Btu/ft²h. Burn-up of UO_2 in the rod examined was approximately

(*Continued at foot of following page*)

Table 1. Liquid waste before dilution

Type	Average gal/day	μc/day	μc/ml
	During Operation		
Non-active waste	1 830	5·7	$8·0\times10^{-7}$
Special liquid waste	1 270	28·2	$5·9\times10^{-6}$
Reactor plant liquid waste	1 820	66·5	$9·6\times10^{-6}$
Chemical waste	151	4·5	$7·8\times10^{-6}$
	During Refuelling		
Non-active waste	2 900	36	$3·3\times10^{-6}$
Special liquid waste	4 300	1 000	$6·1\times10^{-3}$

AEI RUGBY RESEARCH LABORATORIES

THIRTY-SEVEN YEARS after the formation of a Research Section at the Rugby works of the British Thomson-Houston Co., Ltd., the latest laboratories of what is now AEI (Rugby), Ltd., were open to inspection by visiting engineers and scientists as an up-to-date example of modern laboratory design. The building, 370 ft long, has a floor area of 45 000 ft² and, in addition to library, lecture theatre, conference rooms and offices, will provide accommodation for many of the "outlying" projects which have hitherto been housed in temporary premises since the previous laboratories were outgrown.

Meticulously planned on a modular system, the laboratory accommodation is exceedingly flexible and, while there is not the slightest sign of the "temporary" about the partition walls, which are particularly solid and soundproof, it is possible to arrange small or large rooms at short notice, without introducing problems of lighting, heating, ventilation, or easy access to essential services.

General view of the new laboratories.

The Nuclear Field

The major portion of AEI's work directly in the nuclear field is, of course, carried out elsewhere and a considerable amount of the work seen in progress was of marginal interest—at the moment, since many projects, such as direct thermo-electric generation, for example, will undoubtedly concern the nuclear field in the future. On the other hand several projects, such as the refining of metals by electron bombardment (a 10-kW vacuum furnace for the U.K.AEA is practically complete) are of considerable direct interest to the nuclear industry. A particularly noteworthy item in the metallurgical section was the results obtained on research into creep-resistant magnesium alloys for fuel element canning. The "Magzan" series of ternary alloys of magnesium with zirconium and manganese, are giving very satisfactory creep results at 500°C, with fine grain size.

(*Continued from previous page*)

$1{\cdot}77\times10^{20}$ fissions/cm³; or about 7 000 MWd/t of natural uranium.

Underwater examination of the seed elements showed no significant swelling or mechanical distortion. Peak burn-up was about 68% of U^{235} or about 6×10^{20} atoms of U^{235} fissioned/cm³ of fuel alloy.

Water Disposal

The maximum permissible level of waste discharged to the Ohio River (exclusive of tritium) is less than 10^{-6} μc/ml. and liquid waste is usually diluted by 10^{5}-10^{5} by controlled discharge to the condenser outlet. As a result of the high integrity of fuel elements, quantities of waste were less than anticipated; the quantities after processing but before dilution are given in Table 1.

The total amount of gaseous waste discharged from December, 1957, to the end of September, 1959, was 5 157 ft³ containing 0·4 c of activity. Solid waste (incinerator ash, glass, tools, etc.) shipped for sea burial during the same period amounted to 146 55-gal drums, each containing one-third waste and two-thirds cement.

Activity Levels

In general, activity levels have been lower than predicted. During the first few months of operation, full-power levels showed a slight increase, afterwards reaching steady-state values. Shut-down intensities, on the other hand, show a steady increase, due to build-up of corrosion products. Radiation intensities in different parts of the plant are summarized in Table 2.

Similar information is given for SM-1 (originally the APPR) SL-1 (up to the time of the accident) BORAX, OMRE, EBR, and SRE.

Table 2. Radiation Intensities (mr/h)

Full-power Operation	
Primary loop inlet to heat exchanger	6 000/14 000
Walkways, 10 ft above line	200/1 000
In aux. chamber*	150/250
Purification valve access cubicles*	0·03/2
Pressurizer compartment	100/400
Shut Down†	
Boiler chambers, general area	2/30
Heat exchangers (contact)	200
Inside heat exchanger plenum	3 000
Purification system regenerative heat exchanger	400/800
Reactor chamber	20/1 200
Refuelling Operation (Reactor pit)	
After removal of reactor dome (near refuelling seal)	25/40
Inside rod drive mechanism array before installation of lead shield	375/1 500
After installation of lead shield	25/600
After removal of position indicator coils, coil housing, stators, and water jacket assemblies on contact with motor tubes	1 000/1 500
At one foot from motor tubes before shielding	500
At one foot from motor tubes after shielding	250
At outer radius of refuelling seal	15
After cutting seal-weld and removal of mechanism motor tubes and rotors:	
Just outside of mechanism array	70/180
At inner edge of seal ring	30/80
Near walls of reactor pit	7/10
During inspection of reactor vessel internals prior to cleaning and flooding for replacement of fuel assemblies:	
Over fuel ports	15/50
At seal ring	3/30
Radiation levels of removed components	
Control rod drive mechanism rotors: Before decontamination—contact reading	150/2 700
After decontamination at Shipping-port‡	100/840
Mechanism motor guide tube	9/50
Thermal barriers	175/900
Seed metal-temperature thermocouples	180/230

* Accessible at all times.

† During the period August 1958 to November 1959, the intensities near the coolant piping increased by as much as six times and on the reactor head by a factor of about four which was due to deposit of activated corrosion products.

‡ An average decontamination factor of 5·3 was obtained with an ultrasonic cleaning technique.

Books Received

Field Theory for Engineers. By Parry Moon and Domina Spencer. (530 pp. D. Van Nostrand Co., Ltd. 96s.)

Control of Nuclear Reactors and Power Plants. By M. A. Schultz. (Second Edition, 462 pp. McGraw-Hill Publishing Co., Ltd. 97s.)

Power Reactor Technology. Edited by James K. Pickard, Frederick H. Warren, William W. Lowe and Stuart McLain. (The Second Geneva Series on the Peaceful Uses of Atomic Energy.) (417 pp. D. Van Nostrand Co., Ltd. 84s. 6d.)

An Introduction to Servomechanisms. By F. L. Westwater and W. A. Waddell. (191 pp. The English Universities Press, Ltd. 15s.)

Modern Physics for the Engineer. Edited by Louis N. Ridenour and William A. Nierenberg. (Second series, 383 pp. McGraw-Hill Publishing Co., Ltd. 74s.)

Buyers' Guide—Nuclear Industry in Germany. Edited by W. D. Müller and R. Hossner. (127 pp. Econ-Verlag GmbH. Dusseldorf. U.S. $3.50.)

Radiation Damage in Solids. By D. S. Billington and J. H. Crawford. (450 pp. Princeton University Press: Oxford University Press. 75s.)

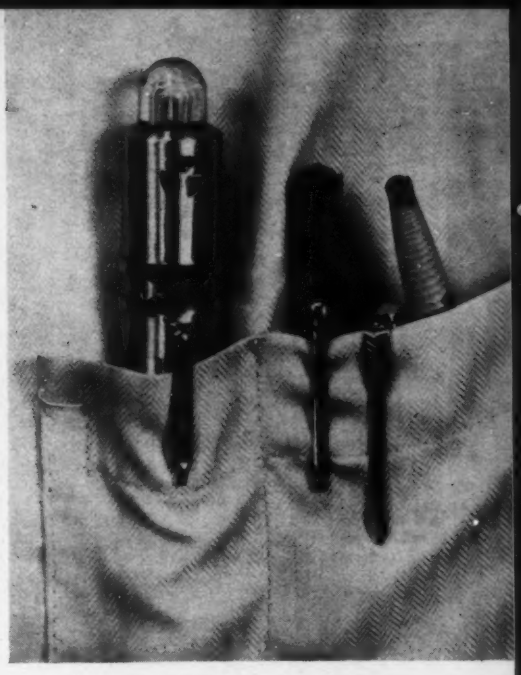
Modern Atomic and Nuclear Physics. By C. Sharp Cook. (296 pp. D. Van Nostrand Co., Ltd. 58s. 6d.)

Rare Metals Handbook; Second Edition. Edited by Clifford A. Hampel. (715 pp. Chapman and Hall. 160s.)

Fuel Element Fabrication, with special emphasis on cladding materials. (Proceedings of IAEA Vienna Symposium, May, 1960.) (538 pp. Academic Press Inc. (London), Ltd. 100s.)

Processes and Equipment

For further information on any item please enter the relevant number on one of the Reader Service Cards enclosed with this issue and forward to the address given.



The NUMEC PRM pocket monitor.

Personal Radiation Monitors

Nuclear Engineering, for July, 1961, contained a tabulated international review of radiation monitoring equipment for personnel protection. Since then, information has been received on other devices worthy of note.

PRM

The PRM, based on a design by ORNL, is a pocket instrument only slightly larger than a fountain pen, being 5¼ in × ¾ in diameter, and weighing 3½ oz. Constructed in stainless steel, and easily decontaminated, it contains a miniature halogen-quenched G-M tube and transistorized circuitry powered by a standard mercury cell, and emits a series of chirping sounds in the presence of radiation; visual indication is simultaneously given by the flashing of a miniature neon bulb. The frequency of the chirps and flashes varies with the gamma radiation and, above certain limits, the chirping sounds merge into a continuous wail. Sounds are emitted near maximum sensitivity of hearing, 2 800 c/s.

Two models are available, the PRM-1 for the range 1 mr/h to 2 r/h, and PRM-2 for 10 mr/h to 200 r/h; the higher figures in each case representing the point of emission of a continuous wail, and the lower level that at which 12/13 flashes and chirps are emitted per minute. More than 1 000 units are being produced by Elgin Laboratories, a division of the Erie Resistor Corporation, for ORNL, and have been tested in fields from background to 4×10^6 r/h without failure.

(NUMEC (Nuclear Materials and Equipment Corporation), Apollo, Pennsylvania.)

1126

SCRAM-1

A similar device, also based on the ORNL design, is being manufactured for REAC (Radiation Equipment and Accessories Corporation) by Radiation Technology Inc. Of similar weight (3½ oz), its dimensions are given as 6 in long × ¾ in diameter. The figures for this device were given in the tabulated review to which reference has already been made; the steady note is given at a level of 1 r/h, while background count gives about one chirp every 2 min.

The REAC SCRAM-1 pocket monitor.

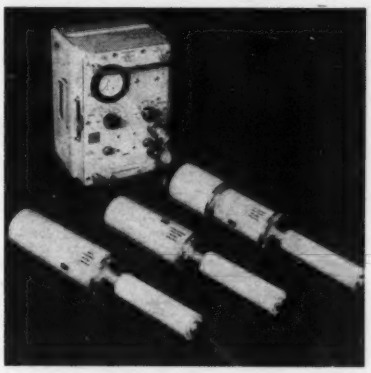
(Radiation Equipment and Accessories Corporation, 665 Merrick Road, Lynbrook, New York.)

1127

Neutron-Gamma Monitor

It is now possible to give further details of the Plessey neutron-gamma monitor Mk. 2NRM, brief particulars of which were given in the July tabulation of personnel monitors.

The 2NRM has been designed to Admiralty specifications for measurements of neutron and gamma radiation in the vicinity of a propulsion reactor. It is a portable instrument using scintillation techniques, with different probes for fast neutrons, thermal neutrons, and gammas.



The Plessey 2NRM neutron-gamma monitor with interchangeable probes.

There are six ranges for thermal neutron measurement, from 0-0·03 to 0-30 mrem/h, and four in the fast neutron scaling (0-1 to 0-300 mrem/h) while gamma radiation measurement is carried out in five ranges from 0-0·3 to 0-300 mrem/h. Gamma radiation for which the instrument has been designed is predominantly in the 4/5 MeV region but the range 0·1-6 MeV is covered. On the neutron ranges, measurements may be made in the presence of gamma radiation of 100 times the equivalent dose-rate (up to a maximum of 300 mr/h).

(Plessey Nucleonics, Ltd., Weedon Road, Northampton.)

1128

Acid Pumps

New Mitchell Craig corrosion-resisting pumps for acids are now being marketed, in capacities up to 200 gal/min, for maximum heads of 120 ft. Two types are being offered, either of the centrifugal type or the positive-displacement diaphragm pattern.

The centrifugal type, has both suction and discharge incorporated in the body, so that the impeller can be withdrawn without having to interfere with the pipework. Either a standard packed gland or a mechanical seal can be supplied; the seal is internally flushed for pumping clear liquors, or an external flushing arrangement can be provided for use with crystal-bearing liquids or slurries. The impeller is of the open type.

Improvements in the diaphragm-type pump include the use of a shorter working stroke, thus increasing the working life of the Hypalon diaphragm, together with a perraux (tulip-type) valve in place of the more usual flap valve, and an integral air bottle to smooth out the flow. Pump internals are of alumina-loaded stoneware, to resist action of all acids except hydrofluoric.

Mitchell Craig acid pumps. Above; centrifugal type. Left; diaphragm type.

(L. A. Mitchell, Ltd., Harvester House, 37 Peter Street, Manchester, 2.)

1129

Proportional Controller

Desirable features of the Pyr-O-Volt 105 R 12 a new Honeywell millivoltmeter instrument for control with magnetic amplifier gear, include straight-line control, automatic fail-safe, built-in voltage regulation and plug-in design, with front access for adjustment and servicing.

Honeywell Pyr-O-Volt proportional controller.

An inexpensive current-proportioning instrument, the Pyr-O-Volt has a continuous output of 3-7 mA working into an impedance of 4 000 ohms or more, when used as the input to a magnetic amplifier, and will control saturable reactors up to 100 kVA.

(Honeywell Controls, Ltd., Greenford, Middlesex.)

1130

Portable Water Purifier

Users of demineralized water, in quantities which do not justify installation of a regenerative plant, can now have their supply literally "on tap." The new "Deminrolit" Mk. 7 has a body of plastics which contains a detachable cartridge containing a bed of intimately mixed Zeo-Carb cation and De-Acidite anion resins, which will provide pure water to B.P. specification, each cartridge being capable of treating 16 gal of London water, or a considerably greater quantity of moorland-type water.

Permutit "Deminrolit" Mk.7 water purifier, with built-in test equipment.

A battery-operated tester and control valves are built into the equipment.

(The Permutit Co., Ltd., Permutit House, Gunnersbury Avenue, London, W.4.)

1131

Rotary Switch Valves

Illustrated is the Drallim multi-way rotary valve, in appearance resembling a rotary switch, suitable for operation on vacuum lines or on pressures up to 200 psi. The design is simple, each outlet being closed by a small poppet valve which opens when a flat on the operating spindle is moved underneath the valve stem. By the use of more than one flat on the spindle, dummy stems and plugs, various "switching" combinations can be obtained. The valves are made with square bodies for 4-way operation or hex. bodies for 6-way working. Double-bank and treble-bank valves are also made. Union nuts (captive seal type) can be supplied for either ¼ in or ⅜ in o.d. metal or plastics tubing.

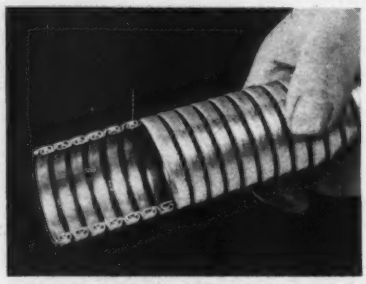
Drallim 6-way single-bank valve.

(Drallim Industries, Ltd., Station Works, Westhall Road, Warlingham, Surrey.)

1132

Flexible Aluminium Tubing

Flexible metallic tubing, hitherto met with in galvanized and stainless steel or bronze, has now been developed in aluminium alloy. The construction, with its interlocked edges and cord-type seal, can be seen from the accompanying illustration. The material used is Noral 54 S (B.S. 1470 NS 5) supplied by Alcan Industries, Ltd. (formerly Northern Aluminium). This tubing will doubtless find many applications where the corrosion-resisting properties and light weight of aluminium are desirable. Pressure limitations are about 300 psi for ¼-in bore tubing to 75 psi for 12-in bore.



Flexible tubing in Noral aluminium alloy.

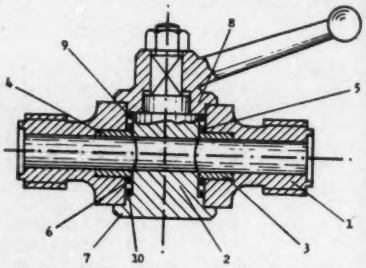
(United Flexible Metallic Tubing Co., Ltd., Kingsbourne House, High Holborn, London, W.C.1.)

1133

Rotary Plug Valves

The Orseal valve, developed by Threlfall, is a plug valve with a difference—it does not depend on ground surfaces for valve tightness, or on packed glands for spindle leakage.

As can be seen from the sectional drawing, there is a considerable clearance between the plug and the valve body, sealing being obtained by O-rings within this clearance space, surrounding the spigots projecting into the body. Additional O-rings seal the top and bottom of the plug, eliminating any need for packed glands. For use on vacuum lines, these plug sealing rings are doubtless find many applications where the corrosion-resisting properties and light weight of aluminium are desirable. Pressure limitations are about 300 psi for ¼-in bore tubing to 75 psi for 12-in bore.



Section of "Orseal" valve. 1, body; 2, plug; 3,4, spigots; 5,6, bore-sealing O-rings; 7, plug collar; 8, loose collar; 9,10, gland O-rings.

BRIEFLY . . .

Claude Lyons, Ltd., Valley Works, Hoddesdon, Herts., have developed a new high-current stabilized d.c. power supply, the type PSS-32/50, suitable for 50 A output at 24-32 V, with stability, after correction of ±0.5%, and 1% rms ripple. A change in a.c. mains voltage of 20 V will be corrected in less than 1 second. Dimensions are 29⅞ in×19⅝ in ×20⅝ in (759×492×524 mm) and weight 322 lb (146 kg). **B284**

Rocol, Ltd., Rocol House, Swillington, Nr. Leeds, are marketing Molytherm, a molybdenum disulphide-containing grease, suitable for lubrication up to 350°F (177°C) continuously or 550°F (288°C) for short periods. **B285**

New developments by **Kabi (Electrical and Plastics), Ltd.,** Kabi Works, Potters Bar, Middlesex, include 5 and 10 gallon covered bins in black polythene. **B286**

Semtex, Ltd., 19/20 Berners Street, London, W.1, a member of the Dunlop Group, is marketing Foamflex a lightweight flexible pipe insulation suitable for operation from −50°C to 130°C, with a thermal conductivity of 0.22 Btu/in ft² °F h. Supplied in lengths of 9 ft 9 in, slit down one side for rapid installation, it is made for pipes of ⅜, ½, ¾, 1, 1¼, 1½, 2, 2½, 3, 3½ and 4 in nominal bore. **B287**

A slide rule, incorporating all the information in their catalogue is available from **British Manufactured Bearings Co., Ltd.,** of Crawley, Sussex. Data incorporated includes the range of ball sizes, grades and tolerances, inner and outer ring tolerances, etc., in inch and metric sizes, for the complete range of BMB ball races. **B288**

A recent development of the **Cambridge Instrument Co., Ltd.,** 13 Grosvenor Place, London, S.W.1, is an electrochemical dissolved oxygen analyser, Mk. IV, for accurate and continuous measurement of feedwater. The gas transfer technique, physically transferring the oxygen from the sample to the measuring cell by scrubbing with hydrogen, enables measurements to be made down to 0.5 ppm. **B289**

A further Cambridge development is the dissolved hydrogen analyser. In this, the hydrogen is scrubbed with oxygen and analysed by a katharometer which has two platinum filaments, one sealed in pure oxygen and the other exposed to the hydrogen-containing sample gas, the two filaments being balanced in a bridge circuit. Concentrations in the range 0-0.02 ppm can be detected. **B290**

Recent developments by the Fluorocarbons Section of **AEI Radio and Electronic Components Division,** 155 Charing Cross Road, London, W.C.2, include an all-PTFE pump and a pipe coupler. The pump, of the diaphragm type, has a capacity of 2 litres/min at 1 ft head or, at its maximum head of 15 ft, 1.25 l/min. Inlet and outlet ports accommodate the new ⅜ in pipe coupling. This coupling, made in five sizes from ⅜ in to ⅞ in, enables PTFE tubing to be joined (to withstand 50 psi at 100°C) without exposing any other material to the working fluid, and without restricting the flow. **B291**

A recent development announced by **United Coke and Chemicals Co., Ltd.,** P.O. Box 136, Handsworth, Sheffield, 13, is "Orkot" Grade TL, a resin-bonded fabric particularly suitable for operation with aqueous lubricants (although for dry conditions it may be supplied impregnated with graphite or moly. disulphide). Dimensionally stable in the presence of water, it has a high resistance to radiation damage, irradiation of 1 000 Mrads causing only 17% diminution in the edgewise compressive strength (normally 15 000 psi, with a "flatwise" compressive strength of 49 000 psi). It is being used as bearing bushes in the fuel handling plant at both Hinkley Point and Sizewell. **B292**

trapped in grooves, to prevent them being drawn inwards.

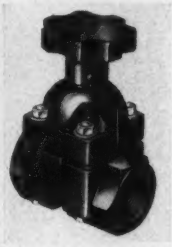
The standard range is ¼ in to 3 in bore; valves can be manufactured up to 12 in bore. Pressures may be as high as 5 000 psi; vacuum up to 10^{-3} mmHg; temperatures from −80°F to 400°F. Flange, screwed, or compression-type ends can be provided; materials include cast iron, bronze, cast steel, stainless steel, aluminium and Fluon. Electrical, pneumatic, hydraulic, etc., operation can be arranged if desired.

(Richard Threlfall, Ltd., Bridgeman Place Works, Bolton, Lancs.)

1134

Corrosion-resistant Valve

When one speaks of a corrosion-resisting valve, one normally thinks of the valve internals, and overlooks the possibility



Saunders corrosion-resistant valve.

of external corrosion of the valve by acid-laden atmospheres in which it may be situated. Illustrated is one of the Saunders type A diaphragm valves in which the body, bonnet and handwheel are in HSB (high styrene butadiene) plastics material, which is similar in chemical resistance to ebonite, but lighter. If required, the bonnet can be in cast iron, or other material. This type of valve is suitable for working pressures up to 80 psi, the maximum temperature being 70°C. It is available in pipe sizes of ¼ in, ½ in and 1 in.

(The Saunders Valve Co., Ltd., Cwmbran, Newport, Mon.)

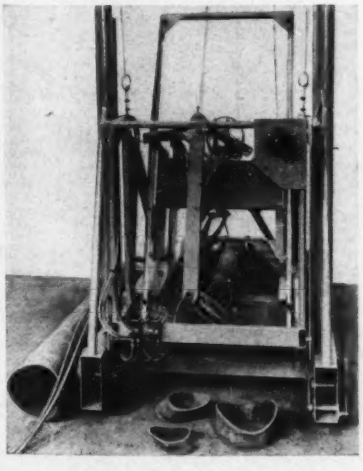
1135

Pipe-profiling Machine

Profiling the ends of pipes which have to be joined into other pipes, particularly when the joint is unsymmetrical or offset, can cause a lot of work in the drawing of sections, preparation of templates, marking-out, hand cutting and trimming.

The Tubefile machine, designed to eliminate this preparatory work, is a profile flame-cutter which will accurately profile the ends

The Tubefile pipe-profiling machine, with some examples of cuts made.



of pipes from simple centre-line drawings, adjustments being provided for the diameters of the respective pipes, the notch angle, offset (if any) and distance of cutter. Multiple profiles (for the common intersection of several tubes) can also be cut. Average accuracy is about 1/64 in.

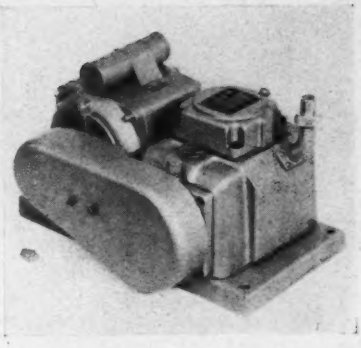
(General Precision Systems, Ltd., Bicester Road, Aylesbury, Bucks.)

1136

Rotary Vacuum Pumps

N.G.N. two-stage rotary vacuum pumps have undergone a considerable amount of redesign to incorporate features which give improved pumping performance and provide facilities for accurately controlling the degree of gas ballast and automatic oil-emptying of the pump casing.

The most important feature of the new design is in the method of connection of the two stages which results in a pump with which the permanent gas ultimate pressures obtained are generally better than 10^{-4} mmHg without gas ballast and better than 10^{-3} mmHg with full gas ballast flow (10% pump capacity). The gas ballast flow is



The N.G.N. two-stage vacuum pump.

controlled by a needle valve which may be locked in any position thus ensuring repeatable pumping performance on continuous cycling with condensable gas loads. Capacities available are 1, 3 and 6 ft³/min.

(N.G.N. Electrical, Ltd., Avenue Parade, Accrington, Lancs.)

1137

CATALOGUES

Spare-parts lists for Mark 3 Multelec electronic instruments have been issued by **George Kent, Ltd.**, Luton, Beds.

Leaflets describing gas analysis records and pneumatic servomotors have been received from **James Gordon and Co., Ltd.**, Dalston Gardens, Stanmore, Middx.

A leaflet describing the P4 Mark II self-priming pump, suitable for 4 000 gallons per hour and capable of pumping water containing 25% solids, has been received from **Ransomes and Rapier, Ltd.**, Waterside Works, Ipswich.

"The Design and Production of Printed Circuits" is the subject of a catalogue produced by **Technograph Electronic Products, Ltd.**, Fleet, Aldershot, Hants.

"Crusilite" heating elements are described in a new publication from **Morganite Electroheat, Ltd.**, Point Pleasant, Wandsworth, London, S.W.18.

"Prestincert," a new type of threaded insert for metal or insulating materials, is described in a catalogue from **Belling and Lee, Ltd.**, Great Cambridge Road, Enfield, Middx. A further publication describes the Collecon and Camlecon connectors.

Mandoval, Ltd., 59 Gresham Street, London, E.C.2, a company under the management of the Rio Tinto Group, have issued a handbook on the properties and applications of vermiculite.

Some of the activities of **Thos. W. Ward, Ltd.**, Albion Works, Sheffield, are reviewed in a booklet "Talking about Wards."

Hydex industrial and laboratory aprons made from a nylon-based pvc material are described in a leaflet received from **Jelteck, Ltd.**, Green Lane, Hounslow, Middx.

"Silicones in Electronics" is the title of a 24-page booklet issued by **Midland Silicones, Ltd.**, 68 Knightsbridge, London, S.W.1.

"Chemicals for Ultrasonic Cleaning" is the title of a booklet issued by the Ultrasonic Power Division of **Branson Instruments, Inc.**, Stamford, Conn., U.S.A.

The Imperial Aluminium Co., Ltd., P.O. Box 216, Witton, Birmingham, 6 (a subsidiary of I.C.I., Ltd.), have issued a 68-page brochure giving the technical data on Impalco aluminium alloys, brazing materials and other products.

Mobile X-ray units, 300-400 kV, are described in a leaflet received from **Pantak, Ltd.**, Vale Road, Windsor, Berks.

New publications from **Turner Bros. Asbestos Co., Ltd.**, Rochdale, Lancs, describe "Durestos" resinated asbestos moulding material and "Duraglas" glass fibre reinforcement.

Araldite for heavy-duty floor service is described in a leaflet issued by **CIBA (A.R.L.), Ltd.**, Duxford, Cambridge.

Some of the activities of **Head, Wrightson and Co., Ltd.**, Ship House, 20 Buckingham Gate, London, S.W.1, are reviewed in a coloured brochure recently issued in four languages.

Holmes-Elex electrical precipitators are described in the latest brochure issued by **W. C. Holmes and Co., Ltd.**, Turnbridge, Huddersfield.

L. Light and Co., Ltd., Poyle Estate, Colnbrook, Bucks, have issued a catalogue of ultra-pure elements and compounds.

The Esab Group of companies, of Gillingham, Kent, have issued a book on welding technique and practice with full technical data, covering the Esab range of OK electrodes.

Bulletins issued by **The Kleen-E-Ze Brush Co., Ltd.** (Industrial Division), Hanham, Bristol, describe "Keza" bullet tube cleaners and "Kez-strip" industrial brushes in metal-backed strip form.

"Engineering in Rubber," a booklet for engineering students, is available from the Education Section, Public Relations Department, **Dunlop Rubber Co., Ltd.**, 10-12 King Street, London, S.W.1.

Negretti and Zambra, Ltd., 122 Regent Street, London, W.1, have issued a booklet giving particulars of overseas sales and service organizations.

From **Isotope Developments, Ltd.**, Bath Road, Beenham, Reading, Berks, come leaflets and specifications describing isotope carriers, lead castles, tritium scintillation counter, timers, ratemeters and scalers.

The July/August issue of *AEI Engineering*, published by **Associated Electrical Industries, Ltd.**, 33 Grosvenor Place, London, S.W.1, contains an article on nuclear reactor control.

The June, 1961, issue of the Wild-Barfield *Heat-Treatment Journal*, published by **Wild-Barfield Electric Furnaces, Ltd.**, Elecfurn Works, Otterspool Way, Watford By-pass, Watford, Herts, contains an article on electron beam welding.

Orbits . . .

BY courtesy of our friends in the Power Reactor Development Co. who, very sportingly, flew us several kilogrammes of Hazards Report so that we got them nearly as quickly as did the U.S.AEC, we have been able to get an up-to-date close-up of the Enrico Fermi fast reactor, and to supplement the short and sketchy piece we published in 1957. It would also—we hoped—enable us to draw some useful comparisons with Dounreay.

Until, of course, we came to do it, when we realized that it was like comparing horses and cows; the only common feature being that they have a leg on each corner. Enrico Fermi has three coolant loops against Dounreay's 24; it has mechanical pumps instead of electromagnetic; the coolant travels upwards in the core instead of downwards. This last appears to cloth-heads like ourselves to be logical—on pump failure, the flow doesn't have to reverse before natural-circulation cooling takes over.

On the other hand, upward coolant flow needs some form of hold-down device for the elements. This is very neatly arranged in the Fermi plant, containing all the gubbins for the control rods. It is difficult to see how a similar device could be applied to Dounreay, which depends on a movable core for its control—something that would be more than a little dicey if upward coolant flow were used. The more one looks at it, in fact, the more the horse-and-cow analogy seems to apply. Let us hope that neither turns out to be a "fair cow" in the Australian sense of the term.

Silvery Stream

One point of comparison of the two plants we can't help making is the (apparently) vast difference in the behaviour of the coolant metal. From eye-witness descriptions, the sodium at the Fermi plant is silvery-bright, after treatment with one (1) cold trap. Dounreay seems to be stuffed with hot and cold traps like raisins in a suet pud, yet they still had endless trouble with oxide. Unofficial reports blame the NaK and potassium for this. We have already queried (February, 1961) the reason for using potassium at all.

Proof of the Pudding . . .

As a celebrated Frenchman once said (in French) criticism is easy, but art is difficult. Dounreay has only had a comparatively short work-out, and the Fermi plant has, so far, not worked at all. When they have both had a good long spell of operation it will be possible to say which are the right approaches and which the wrong—a car with the engine at the back is only "wrong" to the engine-at-the-front school. It is, in fact, highly desirable that both should work. The fuel costs of a successful fast reactor could be very attractive; this is emphasized by an article elsewhere in this issue. Incidentally, isn't it funny that the one thing no one has previously said much about is the desirability of providing the fuel element with a little gas space? Vanadium, niobium . . . everything expensive under the sun has been tried in fast reactor fuel elements. And now we hear that it only wants a bit of 'ole!

Water Music?

In the past few years, hydraulic control has developed to an amazing extent (largely due to the requirements of these dratted flying machines) and has been applied to a great many control schemes which, at one time, could only have been possible by electrical means. Even so, we didn't realize just how sensitive hydraulic controls could be, until a recent visit to the Dowty Group's headquarters at Cheltenham. Here, the sensitivity of their "Moog" electrically controlled hydraulic servo valve was demonstrated, in no uncertain fashion, by coupling the piston shaft of the valve to a diaphragm and feeding the output of a tape recorder into the control circuit, when the diaphragm reproduced the music, at least as well as our own radio set. Electrical engineers, beware! They'll have hydraulic transistors before you know where you are.

"The Merry-go-round Broke Down . . ."

"Reactor Merry-go-round," an editorial in our August issue, recounted the saga of the University reactors and the decisions reached by several bunches of amateur Solomons as to their type and siting. The argument has now been settled, for the time being, by the latest Solomon not merely threatening to divide the baby, but having it thrown away altogether . . . there aint a-goin' to be no reactor nowhere. The late W. S. Gilbert or the translated A. P. Herbert might have made another "Trial by Jury" or "Misleading Cases" out of it . . . we're giving up. It's outside our fighting weight.

Engineersmanship

Bernard Benson's book "Strictly Birdsmanship" has already been noted by this column (May, 1961) and the Benson-Lehner Corporation has now sent us "On Being an Egghead, or Engineersmanship for the Shell of it" which is, if possible, even funnier and more bitingly sarcastic. "Once you've taken a job, don't worry about the company making money; the government takes it away in taxes and gives it back in contracts—this is the fundamental law of feedback"..."Make it look difficult—if it's simple any fool could have done it and it lowers your professional standing" are just two of the many barbed shafts aimed at graduate engineers with too good an opinion of themselves.

On the other hand, with employers haunting the colleges for graduates, can they be blamed for feeling that they must be pretty good? A report from the U.S. states that graduates of the Stevens Institute of Technology are commencing employment at slightly less than the equivalent of £200 per month. In the early '20s, our own starting salary (after five months' search for work) was considerably less than £100 per year.... Still, considering that we still feel that humility is something connected with dew-point temperature, it was probably just as well, wasn't it?

Chu This Over

Deep feelings of no enthusiasm are engendered by the rumour which has reached our little neck of the woods that the CEGB propose to resurrect the Chu to replace the Btu. British steam units, with their lubs per squinch, and their thousand lubs per hour, admittedly are not as neat as metric units, and Fahrenheit is a beastly scale (can you, Sir, look us in the face and swear that you ALWAYS remember when the 32 should be added or subtracted?).

Nevertheless, discounting the national fondness for ancient institutions, will anybody benefit from the change? It will be welcomed with enthusiasm by the publishers of steam tables and entropy charts and by the AEA, who are already converts, if the Dounreay report is typical. It will not, in any way, ease the problems of those who use the metric system, who will merely have to memorize yet another lot of conversion factors. Also, forgive-us-for-mentioning-it-and-don't-look-now, but there is quite a sizeable nation across the Atlantic who still use the Btu, and the Fahrenheit scale. Are we going to hamper ready exchange of ideas with them, just for the fun of it? Or have the CEGB decreed that they should change too? Comment is superfluous—but you're going to get it, anyway:—

Most European States agree
To use the handy Calorie
(The gramme and Centigrade degree)
For thermal computation:
But Englishmen—ah, there's the rub—
Avoid this cartel, ring, or club,
And stick to Fahrenheit and Lub
Producing complication.

(It's not for fear of losing face . . .
We're just a very backward race.)

Our scientific boys are quite
Averse to using Fahrenheit
A pressure-group puts up a fight
The C-H-U to sell:
(A Thing of doubtful pedigree)
"You keep your Lubs, BUT" . . . they decree
"You MUST exchange the F for C
And all shall yet be well."

(Forgetting, in their narrowed view,
Americans are backward, too.)

The purists, then, who so advise
Put up a black—to their surprise
The Glorious British Compromise
On which they bet their shirts
For neither party seems to do
In fact, both take the dimmest view
The French . . . "*A bas la Say-Arsh-Oo!*"
The U.S., "Nix!" . . . or "Nerts!"

(No; no one wants the C-H-U
Except, of course, the You Know Who.)

Tangent